NUREG/IA-0413

International Agreement Report

Assessment of TRACE 5.0 Against ROSA Test 3-1, Cold Leg SBLOCA

Prepared by:
S. Gallardo, V. Abella, G. Verdú and A. Querol

Universitat Politécnica de Valéncia
Camino Vera s/n
46022 Valencia, SPAIN

A. Calvo, NRC Project Manager

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Manuscript Completed: March 2012
Date Published: March 2012

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: U.S. Nuclear Regulatory Commission
Office of Administration
Publications Branch
Washington, DC 20555-0001
E-mail: DISTRIBUTION.RESOURCE@NRC.GOV
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

NUREG/IA-0413

International Agreement Report

Assessment of TRACE 5.0 Against ROSA Test 3-1, Cold Leg SBLOCA

Prepared by:
S. Gallardo, V. Abella, G. Verdú and A. Querol

Universitat Politécnica de Valéncia
Camino Vera s/n
46022 Valencia, SPAIN

A. Calvo, NRC Project Manager

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Manuscript Completed: March 2012
Date Published: March 2012

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

ABSTRACT

The purpose of this work is to overview the results provided by the simulation of a cold leg Small Break Loss-Of-Coolant Accident (SBLOCA) under the assumption of total failure of high pressure injection system in the Large Scale Test Facility (LSTF) via the thermal-hydraulic code TRACE5. The work is developed in the frame of OECD/NEA ROSA Project Test 3-1 (SB-CL-38 in JAEA). Test 3-1 simulated a PWR high-power natural circulation due to failure of scram during cold leg SBLOCA with a break size equivalent to 0.1% of the cold leg under assumption of total failure of High Injection System.

A detailed model of the plant and the chronology of events, following these assumptions, has been developed with TRACE5. A comparison of the simulation of the test with the experimental results is provided throughout several graphs. An acceptable general behaviour is observed in the entire transient. In conclusion, this work represents a good contribution for assessment of the predictability of computer codes such as TRACE5.

FOREWORD

Extensive knowledge and techniques have been produced and made available in the field of thermal-hydraulic responses during reactor transients and accidents, and major system computer codes have achieved a high degree of maturity through extensive qualification, assessment and validation processes. Best-estimate analysis methods are increasingly used in licensing, replacing the traditional conservative approaches. Such methods include an assessment of the uncertainty of their results that must be taken into account when the safety acceptance criteria for the licensing analysis are verified.

Traditional agreements between the Nuclear Regulatory Commission of the United States of America (USNRC) and the Consejo de Seguridad Nuclear of Spain (CSN) in the area of nuclear safety research have given access to CSN to the NRC-developed best estimate thermalhydraulic codes RELAP5, TRAC-P, TRAC-B, and currently TRACE. These complex tools, suitable state-of-the-art application of current two-phase flow fluid mechanics techniques to light water nuclear power plants, allow a realistic representation and simulation of thermalhydraulic phenomena at normal and incidental operation of NPP. Owe to the huge required resources, qualification of these codes have been performed through international cooperation programs. USNRC CAMP program (Code Applications and Maintenance Program) represents the international framework for verification and validation of NRC TH codes, allowing to:

- Share experience on code errors and inadequacies, cooperating in resolution of deficiencies and maintaining a single, internationally recognized code version.
- Share user experience on code scaling, applicability, and uncertainty studies.
- Share a well documented code assessment data base.
- Share experience on full scale power plant safety-related analyses performed with codes (analyses of operating reactors, advanced light water reactors, transients, risk-dominant sequences, and accident management and operator procedures-related studies).
- Maintain and improve user expertise and guidelines for code applications.

Since 1984, when the first LOFT agreement was settled down, CSN has been promoting coordinated joint efforts with Spanish organizations, such as UNESA (the association of Spanish electric energy industry) as well as universities and engineering companies, in the aim of assimilating, applying, improving and helping the international community in the validation of these TH simulation codes[1], within different periods of the associated national programs (e.g., CAMP-España). As a result of these actions, there is currently in Spain a good collection of productive plant models as well as a good selection of national experts in the application of TH simulation tools, with adequate TH knowledge and suitable experience on their use.

Many experimental facilities have contributed to the today's availability of a large thermal-hydraulic database (both separated and integral effect tests). However there is continued need for additional experimental work and code development and verification, in areas where no emphasis have been made along the past. On the basis of the SESAR/FAP[2] reports "Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk" (SESAR/FAP, 2001) and its 2007 updated version "Support Facilities for Existing and Advanced Reactors (SFEAR) NEA/CSNI/R(2007)6", CSNI is promoting since 2001 several collaborative international actions in the area of experimental TH research. These reports presented some findings and recommendations to the CSNI, to sustain an adequate level of research, identifying a number of experimental facilities and programmes of potential interest for present or future international collaboration within the safety community during the coming decade.

CSN, as Spanish representative in CSNI, is involved in some of these research activities, helping in this international support of facilities and in the establishment of a large network of international collaborations. In

1 It's worth to note the emphasis made in the application to actual NPP incidents.
2 SESAR/FAP is the Senior Group of Experts on Nuclear Safety Research Facilities and Programmes of NEA Committee on the Safety of Nuclear Installations (CSNI).

the TH framework, most of these actions are either covering not enough investigated safety issues and phenomena (e.g., boron dilution, low power and shutdown conditions), or enlarging code validation and qualification data bases incorporating new information (e.g., multi-dimensional aspects, non-condensable gas effects). In particular, CSN is currently participating in the PKL and ROSA programmes.

The PKL is an important integral test facility operated by of AREVA-NP in Erlangen (Germany), and designed to investigate thermal-hydraulic response of a four-loop Siemens designed PWR. Experiments performed during the PKL/OECD program have been focused on the issues:

- Boron dilution events after small-break loss of coolant accidents.
- Loss of residual heat removal during mid-loop operation (both with closed and open reactor coolant system.

ROSA/LSTF of Japan Atomic Energy Research Institute (JAERI) is an integral test facility designed to simulate a 1100 MWe four-loop Westinghouse-type PWR, by two loops at full-height and 1/48 volumetric scaling to better simulate thermal-hydraulic responses in large-scale components. The ROSA/OECD project has investigated issues in thermal-hydraulics analyses relevant to water reactor safety, focusing on the verification of models and simulation methods for complex phenomena that can occur during reactor transients and accidents such as:

- Temperature stratification and coolant mixing during ECCS coolant injection
- Water hammer-like phenomena
- ATWS
- Natural circulation with super-heated steam
- Primary cooling through SG depressurization
- Pressure vessel upper-head and bottom break LOCA

This overall CSN involvement in different international TH programmes has outlined the scope of the new period of CAMP-España activities focused on:

- Analysis, simulation and investigation of specific safety aspects of PKL/OECD and ROSA/OECD experiments.
- Analysis of applicability and/or extension of the results and knowledge acquired in these projects to the safety, operation or availability of the Spanish nuclear power plants.

Both objectives are carried out by simulating experiments and plant application with the last available versions of NRC TH codes (RELAP5 and TRACE). A CAMP in-kind contribution is aimed as end result of both types of studies.

Development of these activities, technically and financially supported by CSN, is being carried out by 5 different national research groups (Technical Universities of Madrid, Valencia and Cataluña). On the whole, CSN is seeking to assure and to maintain the capability of the national groups with experience in the thermal hydraulics analysis of accidents of the Spanish nuclear power plants.

Francisco Fernández Moreno, Commissioner
Consejo de Seguridad Nuclear (CSN)

CONTENTS

FIGURES

TABLES

EXECUTIVE SUMMARY

Failure of scram in Small Break Loss-of-Coolant Accident (SBLOCA) scenarios produced in pressurized water reactors (PWR) has been proved to cause high core power for a long time and thermal-hydraulic response that hinders core cooling, gradually decreasing the primary coolant mass inventory. In such scenarios, there is a chance of natural circulation to occur, meeting different conditions as high core power with supercritical flow in the hot legs and counter-current flow limiting (CCFL) at the inlet of the steam generator (SG) U-tubes that holds a large amount of coolant in the tubes. In such context, detailed study of the transition between subcritical and supercritical flows is crucial for reactor safety analysis, being the Froude number (a dimensionless value that describes different flow regimes of open channel flow) an essential parameter to consider.

In this study, developed within the OECD/NEA ROSA Project Test 3-1 (SB-CL-38 in JAEA), a cold leg SBLOCA under the assumption of total failure of high pressure injection system has been studied in the Large Scale Test Facility (LSTF) and simulated via the thermal-hydraulic code TRACE5. The test reproduces PWR high-power natural circulation due to failure of scram during cold leg SBLOCA with a break size equivalent to 1% of the cold leg, under assumption of total failure of high injection system.

TRACE5 model of the plant has been applied, and the chronology of events set to imitate test conditions. Different thermal-hydraulic models have been tested in order to fully comprehend the high-power natural circulation phenomenology, such as the reflood, the CCFL or the critical flux model. Special attention is devoted to Froude number estimation in hot legs due to the prominence of the transition between subcritical and supercritical flows in high power natural circulation phenomenology. Assessment of TRACE5 leads to the conclusion that the code can successfully reproduce complicated conditions of High Power Natural Circulation, during a cold leg SBLOCA. Deviations from the experimental results are seen during the accumulator activation. Despite this fact, TRACE5 offers a good predictability in this type of accident scenarios

ACKNOWLEDGEMENTS

This paper contains findings that were produced within the OECD-NEA ROSA Project. The authors are grateful to the Management Board of the ROSA Project for their consent to this publication, and thank the Spanish Nuclear Regulatory Body (CSN) for the technical and financial support under the agreement STN/1388/05/748.

ABBREVIATIONS

AFW	auxiliary feedwater
AM	accident management
CAMP	Code Assessment and Management Program
CCFL	counter-current flow limiting
CPU	central processing unit
CSN	Consejo de Seguridad Nuclear (Spanish Nuclear Regulatory Commission)
ECCS	emergency core cooling system
FW	feedwater
HPI	high pressure injection
JAEA	Japan Atomic Energy Agency
kg	kilogram(s)
LSTF	Large Scale Test Facility
l/s	liter per second
m	meter(s)
mm	millimeter(s)
MPa	megapascal
kg/cm^2	kilogram per square centimeter
ºC	degrees Celsius
ºK	degrees Kelvin
MFW	main feedwater
MSIV	main steam isolation valve
MW	megawatt(s)
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NT	normalized to the total time of the transient
NV	normalized to the steady state value
PCT	peak cladding temperature
PV	pressure vessel
PWR	pressurized water reactor
PZR	pressurizer
RV	relief valve
s	second(s)
SBLOCA	small break loss-of-coolant accident
SG	steam generator
SNAP	symbolic nuclear analysis package
TRACE	TRAC/RELAP Advanced Computational Engine

1 INTRODUCTION

Different Small Break Loss-of-Coolant Accident (SBLOCA) scenarios in a PWR have proved failure of scram to cause high core power for a long time and thermal-hydraulic response that hinders core cooling, gradually decreasing the primary coolant mass inventory. In such scenarios, natural circulation can occur gathering different conditions as high core power with supercritical flow in the hot legs and Counter-Current Flow limiting (CCFL) at the inlet of the Steam Generator (SG) U-tubes that hold a large amount of coolant in the tubes.

Heat transfer via steam generators is mainly achieved by two-phase natural circulation after the beginning of the SBLOCA when the reactor coolant system is filled with sufficient water inventory. When the primary coolant inventory decreases, natural circulation ends and steam begins to be condensed in steam generator tubes and flows back to the reactor vessel through the hot legs. This is known as reflux condensation phenomenon. Reflux condensation and natural circulation become important in heat removal when the break size is 0.1% or lower [1]. Actually, the reflux condensation is one of the effective heat removal mechanisms during a SBLOCA. TRACE contains a unified heat transfer package that includes models for reflood heat transfer, which can be activated when reflood is expected to occur.

The supply of cooling water into the reactor core is limited by the CCFL at the reactor vessel downcomer, pressurizer surge line and hot leg pipe. When CCFL occurs under the reflux condensation, it means that the pressure drop across the hot leg increases. This causes pressure increase in the upper plenum of the reactor vessel and the decrease of the liquid level in the reactor core, which results in an increase of the fuel temperature. Due to this, development of CCFL correlation is important for the prediction of system behavior and thermal-hydraulic phenomena.
A detailed study of the transition between subcritical and supercritical flows is also important for reactor safety analysis. In this frame, the Froude number is an important parameter to be considered. Froude Number is defined as the ratio between the mean flow velocity and the speed of an elementary gravity wave travelling over the water surface. Such ratio discriminates subcritical or supercritical flow depending on whether its velocity is higher or lower than the speed of a disturbance wave. There is extensive literature about Froude Number and heat transfer regimes [2, 3, 4].

In this work, the purpose of authors is to describe the most relevant results achieved by using the thermal-hydraulic code TRACE5 in the frame of OECD/NEA ROSA Project Test 3-1 (SB-CL-38 in JAEA) [5, 6], to validate the code predictability. The experiment simulates a high-power natural circulation during 1% cold leg diameter SBLOCA, also considering the failure of the High Pressure Injection System (HPIS). The Test 3-1 of the OECD/NEA ROSA project was handled during 5th of December 2006 in the Large Scale Test Facility (LSTF) of the Japanese Atomic Energy Agency (JAEA). The LSTF [7] simulates a PWR reactor, Westinghouse type, of four loops and 3423 MW of thermal power, scaled to 1/48 in volume and two loops.

In this frame, TRACE5 code [8, 9] has been used in order to simulate a SBLOCA scenario, allowing testing its behavior for high power natural circulation. In addition, different thermal-hydraulic models have been tested in order to fully comprehend the high-power natural circulation phenomenology, such as the reflood, the CCFL or the critical flux model, giving a feedback for developers and programmers.

2 ROSA FACILITY DESCRIPTION

In this section, a brief description of the LSTF facility (in the Tokai Research Establishment of the JAERI) [7] is presented. The primary coolant system consists of the Pressure Vessel (PV), the primary loop A with the pressurizer (PZR) and the symmetrical primary loop B. Each loop contains a primary Coolant Pump (PC) and a Steam Generator (SG). The secondary-coolant system consists of the jet condenser (JC), the Feedwater Pump (PF), the Auxiliary Feedwater Pumps (PA) and related piping system in addition to two SG secondary systems. The ECCSs consist of the high pressure charging pump (PJ), the high pressure injection pump (PL), the Residual Heat Removal (RHR) system and the Primary Gravity Injection Tank (PGIT). The coolant discharged from the primary system is stored in the break flow Storage Tank (ST). The PV is composed of an upper head above the upper core support plate, the upper plenum between the upper core support plate and the upper core plate, the core, the lower plenum and the downcomer annulus region surrounding the core and upper plenum. LSTF vessel has 8 upper head spray nozzles (of 3.4 mm inner-diameter). 8 Control Rod Guide Tubes (CRGTs) form the flow path between the upper head and upper plenum. The maximum core power of the LSTF is limited to 10 MW which corresponds to 14% of the volumetrically scaled PWR core power and is sufficiently capable to simulate PWR decay heat power after the reactor scram.

Regarding SG, each of them contains 141 U-tubes which can be classified in different groups depending on their length (an average length of 19.7 m can be considered, with a maximum height of 10.62 m and a minimum height of 9.156 m). U-tubes have an inner diameter of 19.6 mm and an outer diameter of 25.4 mm (with 2.9 mm wall thickness). As a consequence, the total inner and outer surface areas are 171 and 222 m^2, respectively. On the other hand, vessel, plenum and riser of steam generators have an inner height of 19.840, 1.183 and 17.827 m, respectively. The downcomer is 14.101 m in height.

3 TRANSIENT DESCRIPTION

The break unit is connected to the cold leg inner surface, and the orifice flow area corresponds to 1% of the volumetrically-scaled cross-sectional area of the reference PWR cold leg. The transient starts at time zero with opening the break valve and increasing the rotational speed of the coolant pumps. The complete control logic of the transient is listed in Table 1.

Few seconds afterwards, the scram signal is generated. This signal produces the initiation of the core power decay curve (Table 2), calculated by considering the stored heat in fuel rods and delayed neutron fission power. The PWR core power decreased due to negative feedback induced mainly by an increase in the core void fraction during the transient.

Table 1 Control logic and sequence of major events in the experiment.

Event	Condition
Break.	Time zero.
Generation of scram signal.	Time = 0.004 NT.
Pressurizer (PZR) heater off.	Generation of scram signal or PZR liquid level below 0.306 NV.
Initiation of core power decay curve simulation.	Generation of scram signal.
Initiation of Primary Coolant Pump coastdown.	Generation of scram signal.
Turbine trip (closure of stop valve).	Generation of scram signal.
Closure of Main Steam Isolation Valve.	Generation of scram signal.
Termination of Main Feedwater.	Generation of scram signal.
Initiation of Auxiliary Feedwater.	Generation of scram signal.
Initiation of accumulator system.	Primary pressure decreases to a determined value.

Table 2 Predetermined core power decay curve (normalized to the Steady State power).

Normalized Time	Normalized Power	Normalized Time	Normalized Power
0	1	0.058	0.3042
0.036	1	0.062	0.2763
0.040	0.8150	0.065	0.2423
0.044	0.5366	0.068	0.2263
0.046	0.4504	0.076	0.2079
0.050	0.3906	0.080	0.2000
0.054	0.3538	0.090	0.1913

Normalized Time	Normalized Power	Normalized Time	Normalized Power
0.100	0.1832	0.600	0.0936
0.120	0.1577	0.800	0.0886
0.160	0.1487	1.000	0.0814
0.190	0.1342		
0.256	0.1238		
0.300	0.1096		
0.400	0.1003		

Scram signal produces the following actions: initiation of core power decay curve, initiation of primary coolant pumps coast down (Table 3), turbine trip, closure of Main Steam Isolation Valves (MSIV), termination of Main Feedwater (MFW) and initiation of Auxiliary Feedwater (AFW). When secondary pressure reaches a determined value, Relief Valve (RV) of SGs starts to open and close in order to maintain pressure almost constant. Finally, when primary pressure drops to a determined value, the accumulator water system is activated.

Table 3 Pumps relative rotational speed (normalized values).

Time Normalized	Relative rotational speed	Time Normalized	Relative rotational speed	Time Normalized	Relative rotational speed
0	1.000	0.006	0.280	0.016	0.125
0.0004	0.850	0.008	0.220	0.018	0.110
0.001	0.730	0.01	0.185	0.020	0.100
0.002	0.540	0.012	0.160	0.050	0.000
0.004	0.370	0.014	0.140	--	---

In order to protect the facility, the core power is automatically decreased by the core protection system when the maximum fuel rod surface temperature excess 873 K, as it can be seen in Table 4.

Table 4 Core protection system logic.

Control of core power to	Maximum fuel rod surface temperature (K)
75%	873
50%	983
25%	903
10%	913
0% (core power trip)	923

4 APPLIED METHOD: TRACE5 MODEL OF ROSA FACILITY

TRACE5 code is designed to perform best-estimate analyses of LOCAs or operational transients. However, it has some limitations in use; for example, those transients in which one expects to observe thermal stratification of the liquid phase in the 1D components cannot be directly modeled. Furthermore, it is not appropriate for modeling situations in which transfer of momentum plays an important role at a localized level [9]. In this work, the LSTF has been modeled with 88 hydraulic components (7 BREAKs, 13 FILLs, 29 PIPEs, 2 PUMPs, 1 PRIZER, 21 TEEs, 14 VALVEs and 1 VESSEL). In order to characterize the heat transfer processes, 48 Heat Structure components (Steam Generator U-tubes, core power, pressurizer heaters and heat losses) have been considered. Figure 1 shows the nodalization of the model using the Symbolic Nuclear Analysis Package software (SNAP) [10].

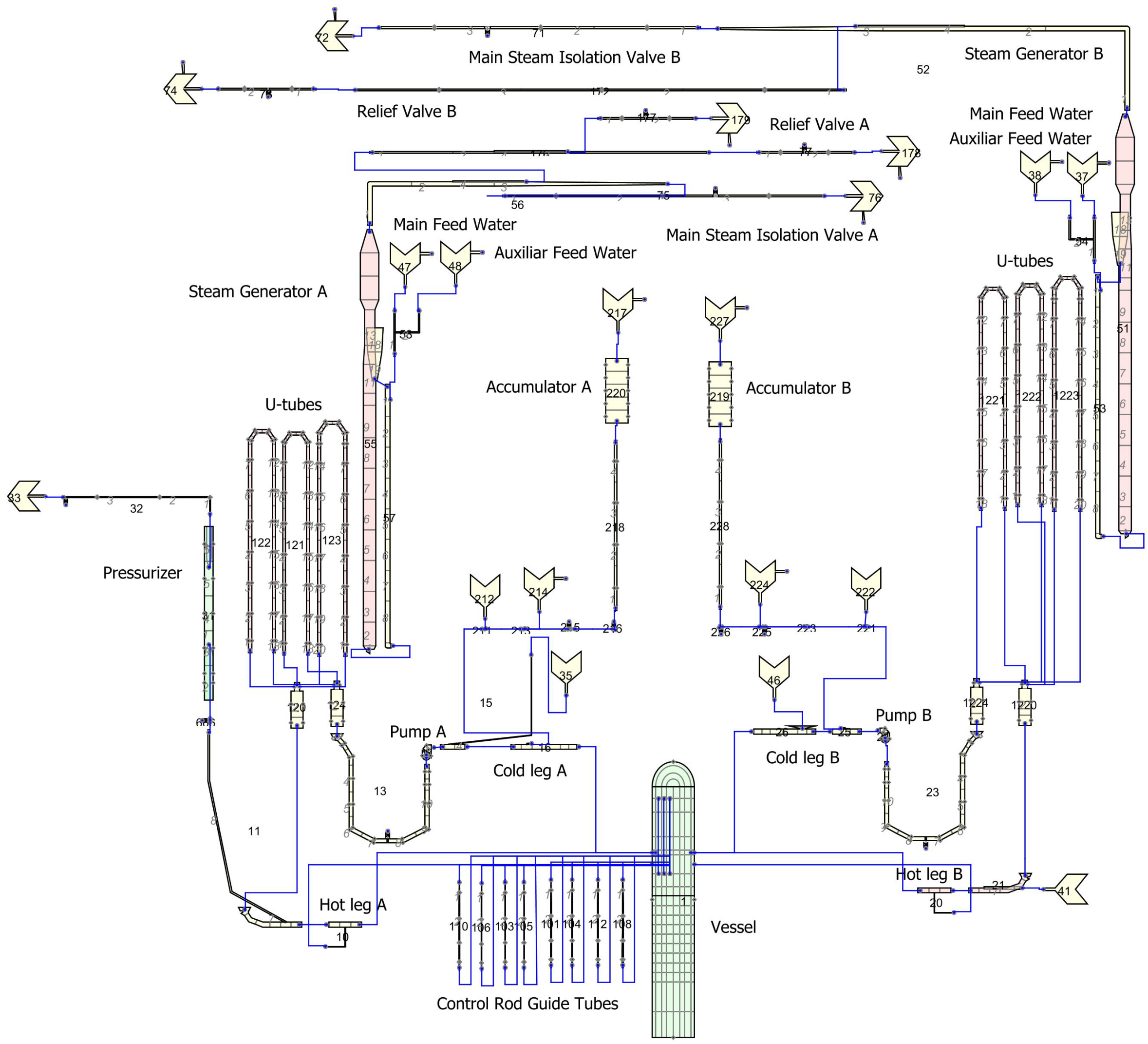

Figure 1 Model nodalization used for simulation.

In order to model the pressure vessel, a 3D–VESSEL component has been considered (Figure 2). A nodalization consisting of 19 axial levels, 4 radial rings and 10 azimuthal sectors has been selected. This nodalization characterizes with an acceptable detail the actual features of the LSTF vessel. Increasing the number of axial levels, azimuthal sectors or radial rings, does not improve significantly the agreement with experimental results, but increases CPU time. For each axial level, volume and effective flow area fractions have been set according to technical specifications provided by the organization [7]. Active core is located between levels 3 and 11. Level 12 simulates the upper core plate. Levels 13 to 15 characterize the vessel upper plenum. The upper core support plate is located in level 16. Finally, upper head is defined between levels 17 to 19. 3D-VESSEL is connected to different 1D components: 8 Control Rod Guide Tubes (CRGT), hot leg A and B (level 15), cold leg A and B (level 15) and a bypass channel (level 14). Control rod guide tubes have been simulated by PIPEs components, connecting levels 13 and 19 and allowing the flow between upper head and upper plenum.

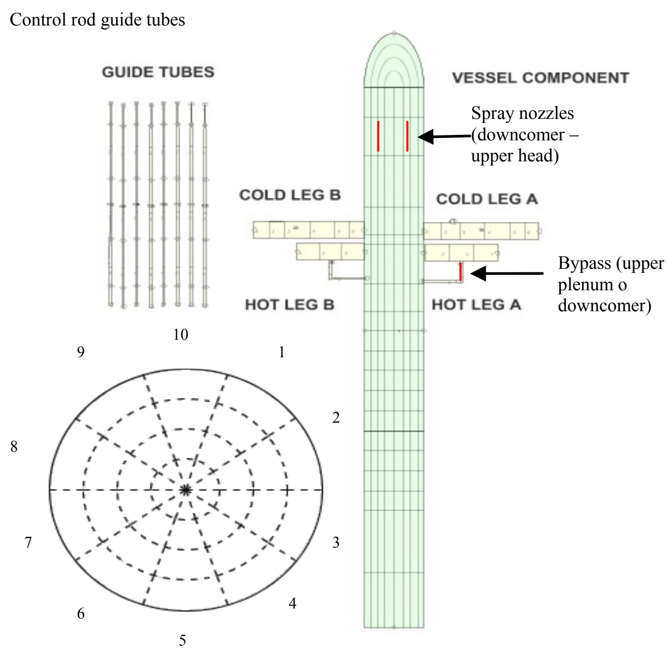

Figure 2 3D Vessel nodalization and connections visualized with SNAP.

30 HTSTRs simulate the fuel assemblies in the active core. The component power manages the power supplied by each HTSTR to the 3D-VESSEL. Fuel elements (1008 in total) were distributed into the 3 rings: 154 elements in ring 1, 356 in ring 2 and 498 in ring 3 and also characterized by HTSTR components. In both axial and radial direction, peaking factors were considered. The power ratio in the axial direction presents a peaking factor of 1.495. On the other hand, depending on the radial ring, different peaking factors were considered (0.66 in ring 1, 1.51 in ring 2 and 1.0 in ring 3). The number of fuel rod components associated with each heat structure has been determined from the technical documentation given, taking into account the distribution of fuel rod elements in the vessel, as it can be seen in Table 5.

Table 5 Number of heaters per heat structure.

HTSTR	Number of heaters	HTSTR	Number of heaters	HTSTR	Number of heaters
310	17	320	44	330	60
311	17	321	40	331	54
312	10	322	23	332	32
313	12	323	32	333	45
314	20	324	40	334	56
315	17	325	42	335	61
316	16	326	38	336	57
317	12	327	26	337	31
318	14	328	30	338	45
319	17	329	39	339	57

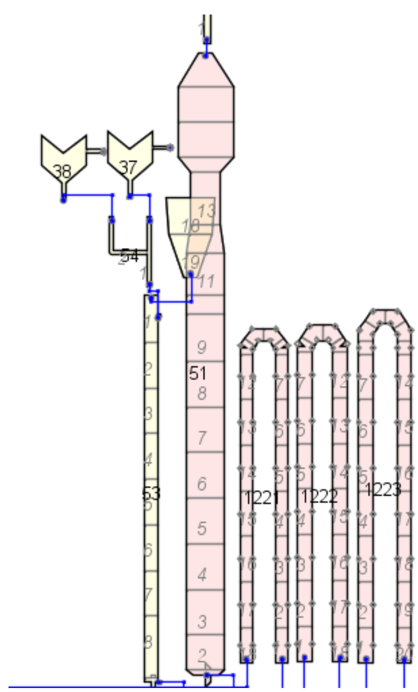

Figure 3 Steam generator nodalization.

A detailed model of SG (geometry and thermal features) has been developed, due to the fact that TRACE5 does not include any pre-determined steam generator component. A representation of the SG nodalization can be seen in Figure 3. Both boiler and downcomer components of secondary-side have been modeled by TEEs components. U-tubes have been classified into three groups according to each average length and heat transfer features. Steam-separator model can be invoked in TRACE5 setting a friction coefficient (FRIC) greater than 10^{22} at a determined cell edge, allowing only gas phase to flow through the cell interface. Heat transfer between primary and secondary sides has been performed by using HTSTR components. Cylindrical-shape geometry has been used to best fit heat transmission. Critical heat flux flag has been set in order to use an AECL-IPPE table, calculating critical quality from Biasi correlation [8, 9]. Inner and outer surface boundary conditions for each axial level have been set to couple HTSTR component to hydro components (primary and secondary fluids). Different models varying the number of U-tube groups were tested (1, 3 and 6 groups). It was found that results do not apparently change, using these models. However, in order to best fit the collapsed liquid level in U-tubes without drastically increasing CPU time, a 3-group configuration was finally chosen. Heat losses to environment have been added to secondary-side walls.

Regarding the break simulation, it is important to take into account the necessity of activating the Choke flow model in the break when critical flow conditions are expected to appear. Choke model predicts for a given cell the conditions for which choked flow is expected to occur, providing three different models in one: subcooled-liquid, two-phase and single-phase vapor model. The break has been simulated by means of a VALVE component connected to a BREAK component in order to establish the boundary conditions. This BREAK has been modelled following the recommendations of the TRACE5 user's manual [9]. In this case, since the break is simulated to discharge in a big volume space (the Storage Tank), a dxin=$1.0*10^{-6}$ (length) and a volin=$1.0*10^{6}$ (volume) has been selected with the purpose of providing a large area.

5 RESULTS AND DISCUSSION

5.1 Steady-state

Steady-state conditions achieved in the simulation were in reasonable agreement with the experimental values. In Table 6, the relative errors (%) between experimental and simulated results for different items are listed. It is important to remark that in any case, the maximum difference between experiment and simulation is 5%. In order to achieve the steady state conditions, the duration of simulation was stated to 3000 seconds.

Table 6 Steady-state condition. Comparison between experiment and TRACE.

Item	Relative Error (%) (Loop with PZR)
Core Power	0.0
Hot leg Fluid Temperature	0.1
Cold leg Fluid Temperature	0.1
Mass Flow Rate	5.0
Pressurizer Pressure	0.2
Pressurizer Liquid Level	4.3
Accumulator System Pressure	0.2
Accumulator System Temperature	0.5
SG Secondary-side Pressure	0.7
SG Secondary-side Liquid Level	3.8
Steam Flow Rate	0.4
Main Feedwater Flow Rate	0.0
Main Feedwater Temperature	0.3
Auxiliary Feedwater Temperature	0.3

5.2 Transient

Table 7 lists the chronology sequence of events during the transient and the comparison in Normalized Time (1 NT = Total Transient Time) between the experiment and TRACE results.

Table 7 Chronological sequence of events. Comparison between experiment and TRACE.

Event	Experiment Normalized time	TRACE5 Normalized time
Valve open	0.000	0.000
Scram signal	0.004	0.008
Initiation of coastdown of primary coolant pumps	0.004	0.008
Primary coolant pumps stopped	0.049	0.052
Termination of continuous opening of SG RVs, termination of two-phase natural circulation, break flow from single-phase liquid to two-phase flow	About 0.054	0.061
Core power decrease by LSTF core protection system	About 0.324	0.325
Max. fuel rod surface temperature	About 0.364	0.400
Primary pressure lower than SG secondary-side pressure	About 0.400	0.405
Initiation of accumulator system (primary pressure lower than a determined value)	About 0.420	0.421
Break valve closure	1.000	1.000

Variables presented in this section follow the requirements for an exhaustive analysis of the transient. These variables include: Pressures in both primary and secondary circuits, mass flow rate and inventory at the break, primary mass flow, vessel collapsed-liquid levels, collapsed-liquid levels in hot and cold legs, Froude number in hot leg, U-tubes liquid level and liquid level in SG secondary-side. Results have been normalized in each graph.

5.3 System pressures

A comparison between primary and secondary pressures is presented in Figure 4 and 5. At time zero primary pressure starts to decrease due to the loss of coolant discharged through the break. The scram signal is generated few seconds after the break, along with the closure of the main Steam Isolation Valves (MSIV) of the Steam Generators (SG), termination of Main Feed Water (MFW), initiation of Auxiliary Feed Water (AFW) and the coast down of the primary coolant pumps. After that, only natural circulation occurs, contributing to cool down the core during the first part of the transient, until 0.06 NT, when the SG Relief Valves (RVs) are fully closed.

Some seconds after the closure of the MSIVs, the secondary pressure of SGs increases reaching its maximum value at 0.05 NT. Simultaneously, primary pressure is recovered temporarily due to MSIVs closure and then starts to decrease along with the beginning of the power decay curve. When secondary pressure reaches a determined value, RV of SGs starts to open and close in order to maintain pressure almost constant. At 0.1 NT, RVs are definitively closed.

The primary pressure becomes lower than the SG secondary-side pressure at about 0.4 NT. The accumulator system is initiated at about 0.42 NT when the primary pressure decreases to a pre-determined value. At this moment, depressurization becomes effective due to steam condensation caused by the coolant injection in the cold legs. Most of the core is quenched at about 0.45 NT after its seal, induced by the condensation of vapour of the coolant injected by the accumulator for the first time, causing a sudden increase of the primary pressure due to the high production of vapour in the core. The primary pressure falls below the secondary pressure at about 0.47 NT after the automatic drop of core power.

In general, both primary and secondary-side pressures are successfully reproduced by TRACE5 in the whole transient, disagreeing only during the accumulator water injection. TRACE5 does not reproduce the small pressure peaks registered during the accumulator water injection.

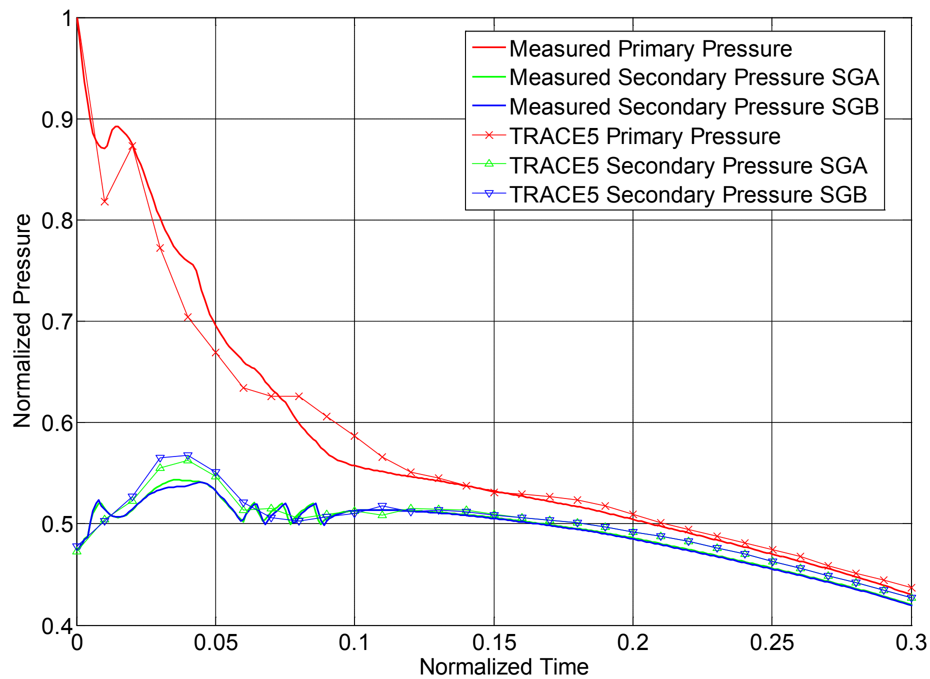

Figure 4 Primary and secondary pressures

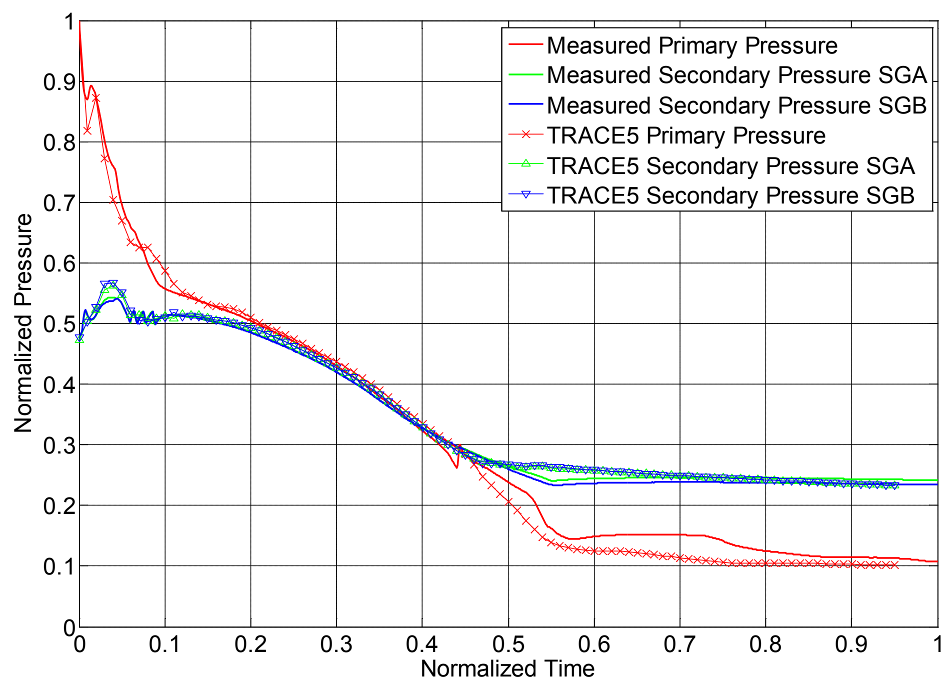

Figure 5 Primary and secondary pressures.

Comparison of pressures between experiment and simulation suggests that condensation of vapour in cold leg produced by the water injection from accumulators is not properly simulated by TRACE5. From this moment on, it can be observed a slight difference in the primary pressure between experiment and simulation till the end of the transient.

5.4 Break

Figure 6 and Figure 7 show the mass flow and the discharged coolant inventory, respectively, through the break. In order to adjust both break mass flow and the discharged primary coolant inventory with TRACE5, a sensitivity analysis varying the discharge coefficient was performed. A discharge coefficient of 1.0 (for single-phase liquid and two-phase), using Choked Flow model [8], fits successfully both mass flow break and discharged coolant inventory, at least until the accumulator water injection (at 0.6 NT), when most part of the coolant was discharged through the break. Steam single-phase coefficient cannot be changed since TRACE5 does not offer this possibility. TRACE5 estimates a change of flow from liquid single-phase to two-phase at around 0.1 NT, which completely agrees with the experimental measurements.

The CCFL model has also been used in the TEE component of the break, following the recommendations, with the default values of the coefficients for the Kutateladze model. A proper adjustment of the discharged mass flow inventory gives a high reliability to the TRACE5 model.

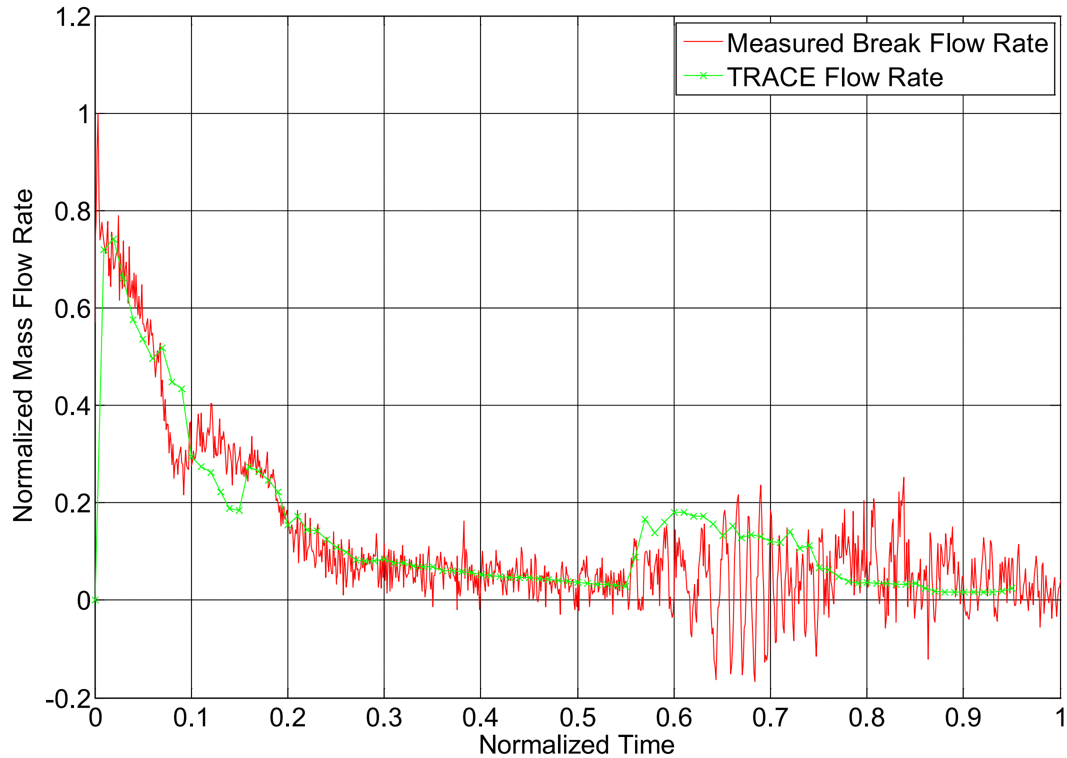

Figure 6 Break mass flow rate.

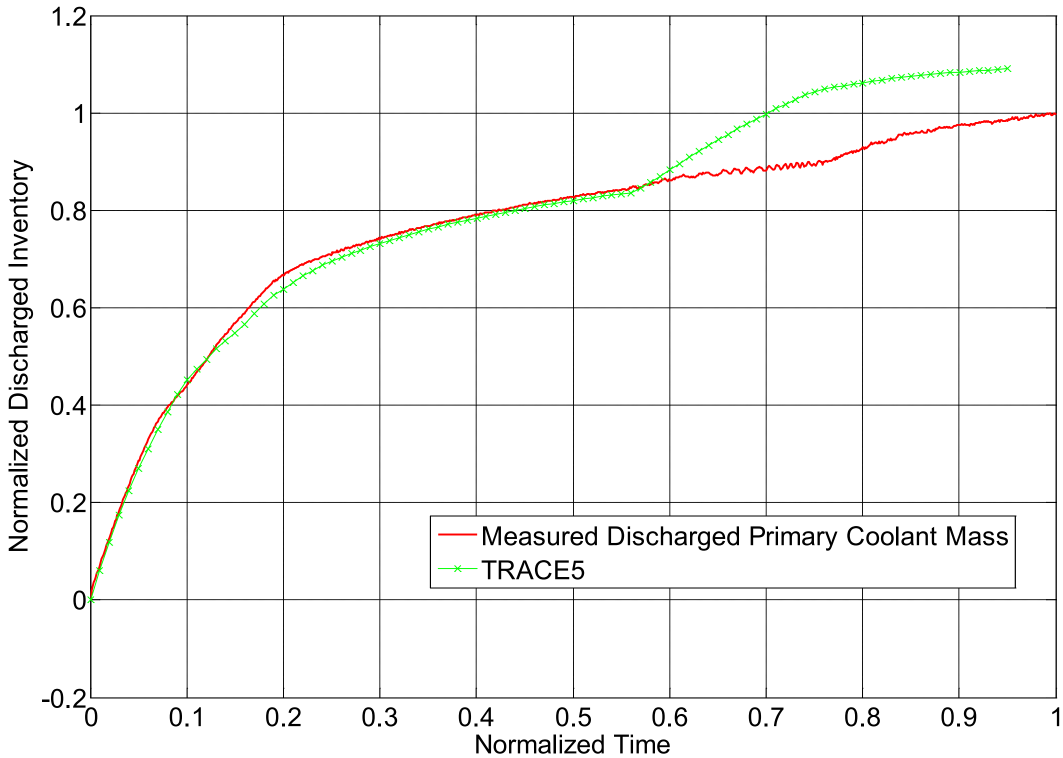

Figure 7 Discharged inventory through the break.

5.5 Primary loop mass flows

Primary loop A and B mass flow rates are shown in Figures 8 and 9, respectively. In the first part of the transient the primary mass flow increases due to the higher angular speed of the pumps. At 0.05 NT it starts to decrease simultaneously with the pumps coast down. After that only natural circulation occurs and it contributes to cool down the core during the first part of the transient, until 0.06 NT, when the SG RVs are fully closed.

Primary loop two-phase natural circulation continued about 0.06 NT when the RV SGs terminated continuous opening. Liquid accumulated in the SG upflow-side U-tubes and inlet plenum during reflux condensation mode. Flow in hot legs became supercritical during two-phase natural circulation due to high vapour and liquid velocity, causing the hot liquid level to be quite low.

TRACE5 results agree with the experimental values. The main discrepancies are found during the accumulator water injection period. As it can be seen in Figure 9, between 0.4, and 0.8 NT, experimental primary mass flow through loop B, increases three times. These increases of mass flow are due to the accumulator water entrance in cold legs. This effect has not been properly predicted by TRACE5.

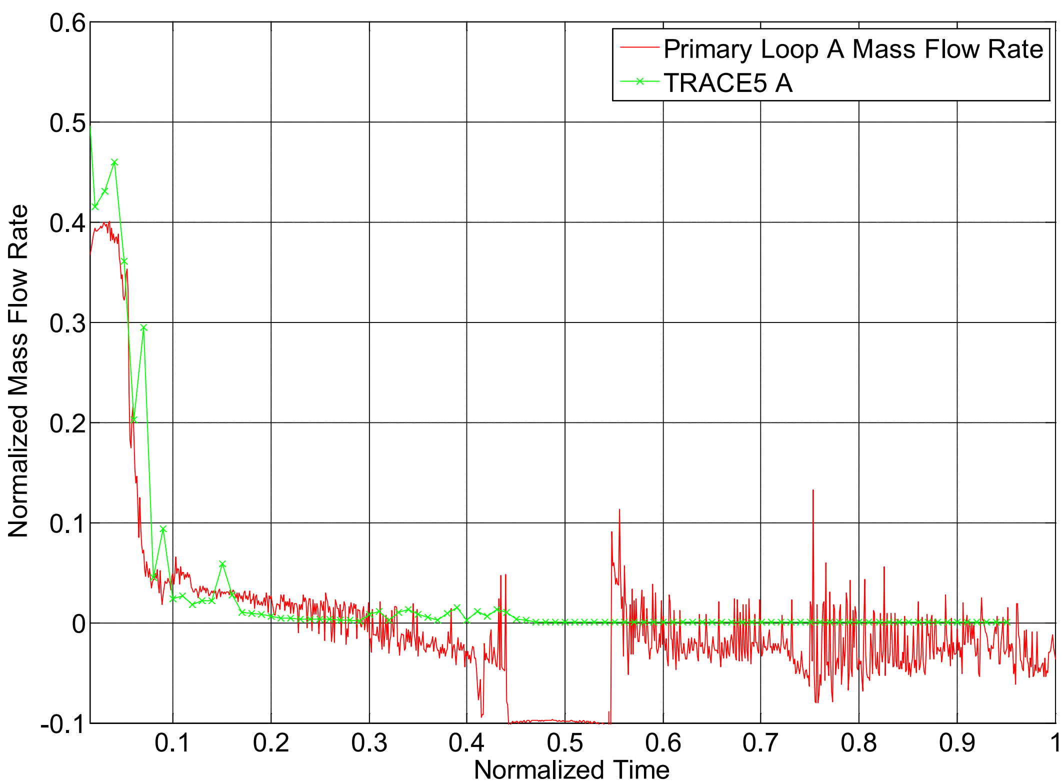

Figure 8 Primary Loop A mass flow rate.

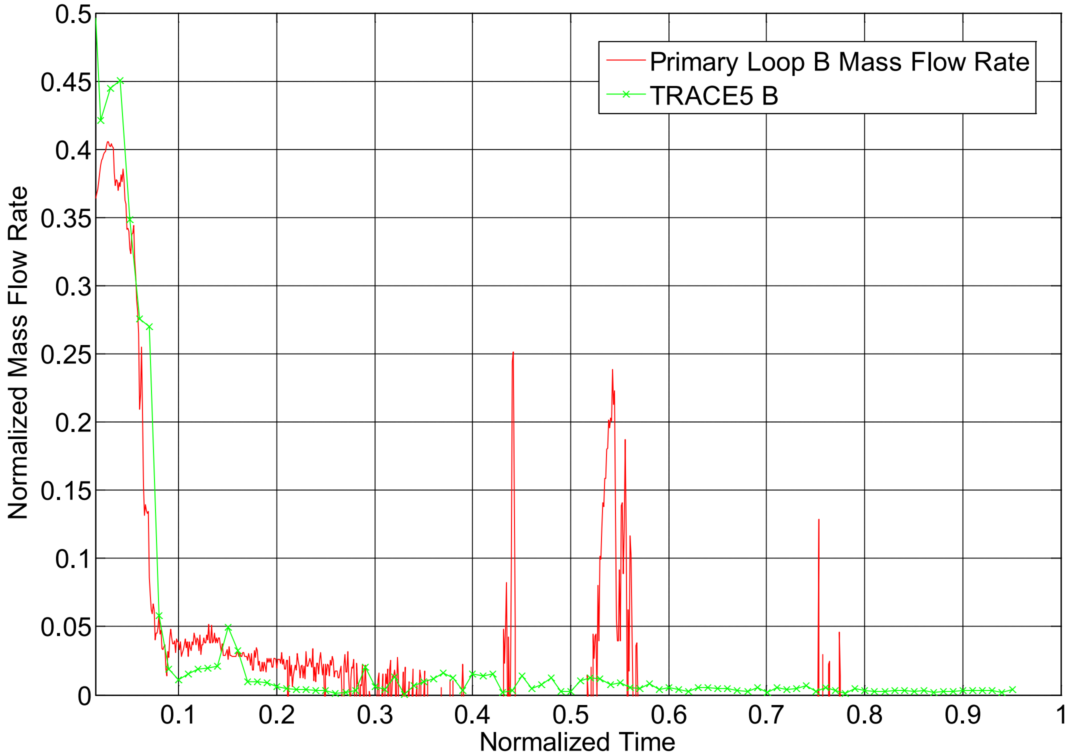

Figure 9 Primary Loop B mass flow rate.

5.6 Vessel collapsed liquid levels

The following Figures (10, 11 and 12) show a comparison between the collapsed liquid levels in the upper plenum, core and downcomer, respectively, for both experimental and TRACE results. All the tendencies are successfully reproduced by TRACE5. Slight discrepancies must be remarked. Core clearance is slightly later simulated by TRACE, even though the core quench is predicted with precision.

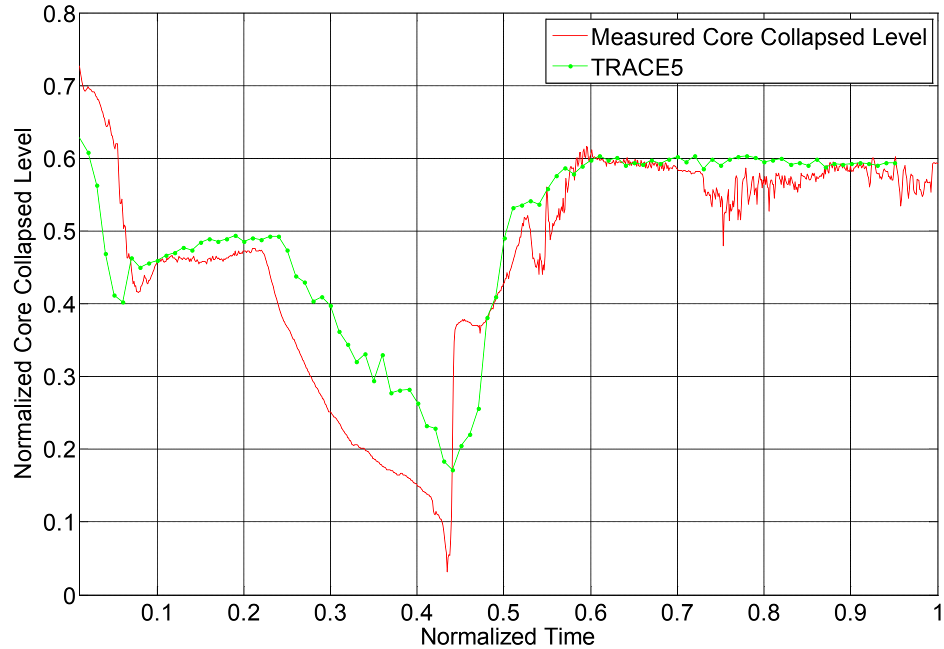

Figure 10 Core collapsed liquid level.

On the other hand, TRACE perfectly simulates the upper plenum clearance and quench.

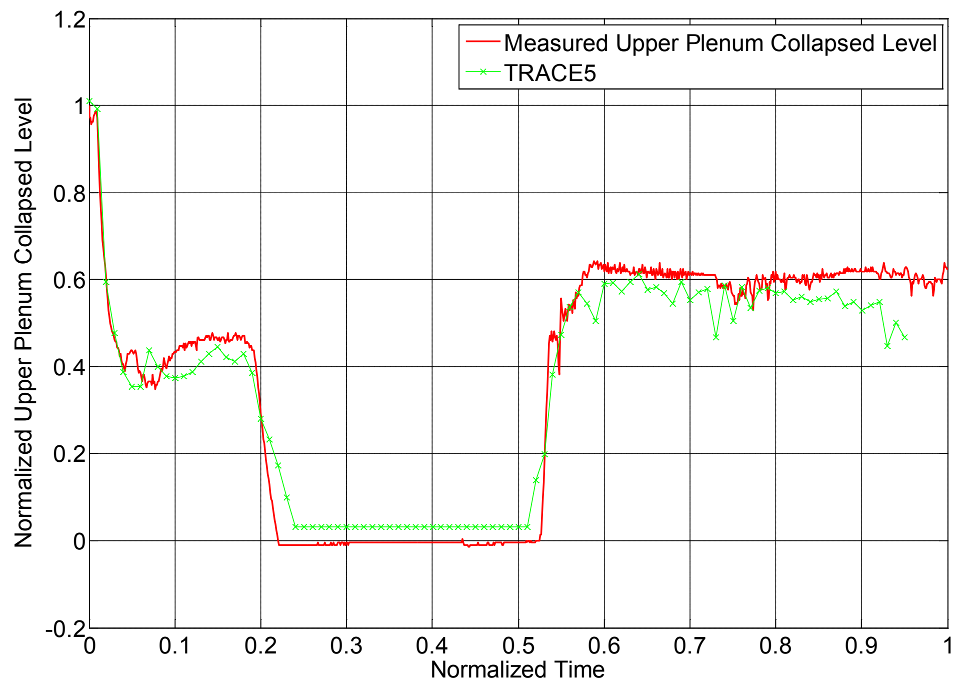

Figure 11 Upper plenum collapsed liquid level.

Concerning the downcomer, TRACE5 also makes a good prediction, only having a slight discrepancy in the first part of the transient, before the clearance, being the level predicted slightly lower than in the experiment.

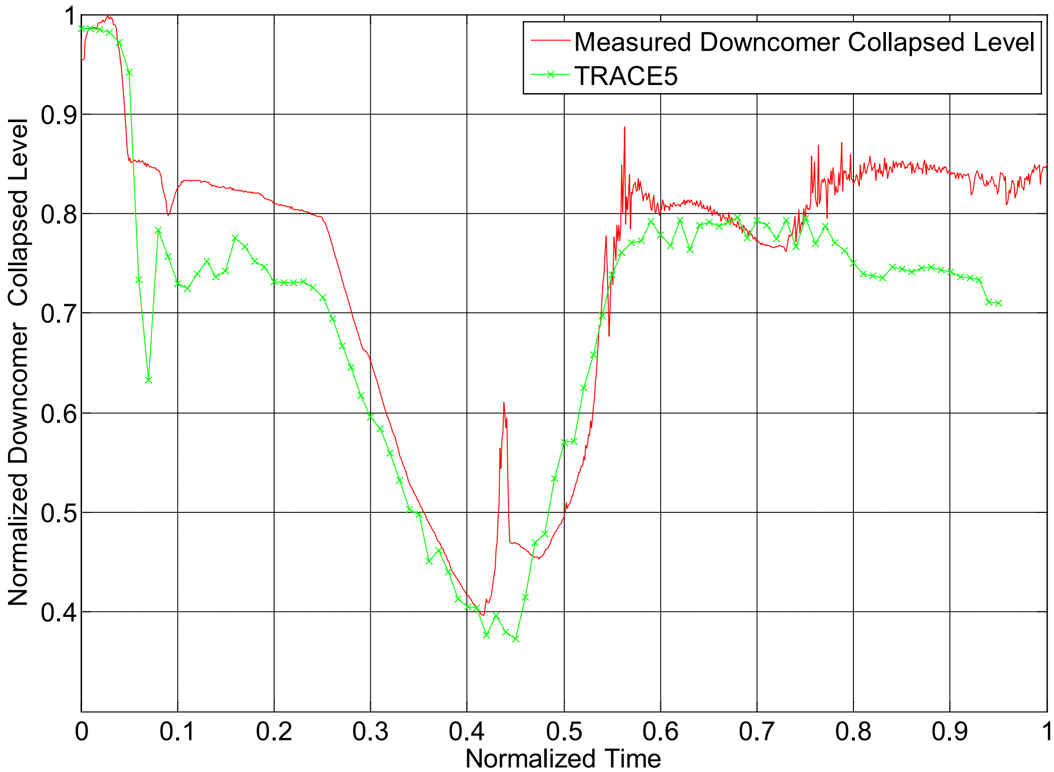

Figure 12 Downcomer collapsed liquid level.

5.7 Maximum fuel rod surface temperature

The maximum rod surface temperature starts to increase when the core clearance takes place. TRACE predicts such increment at 0.24 NT approximately, concurrent with the experiment. At 0.4 NT approximately it reaches the maximum, activating the core protection system by power control, as can be seen in the following Figure 13.

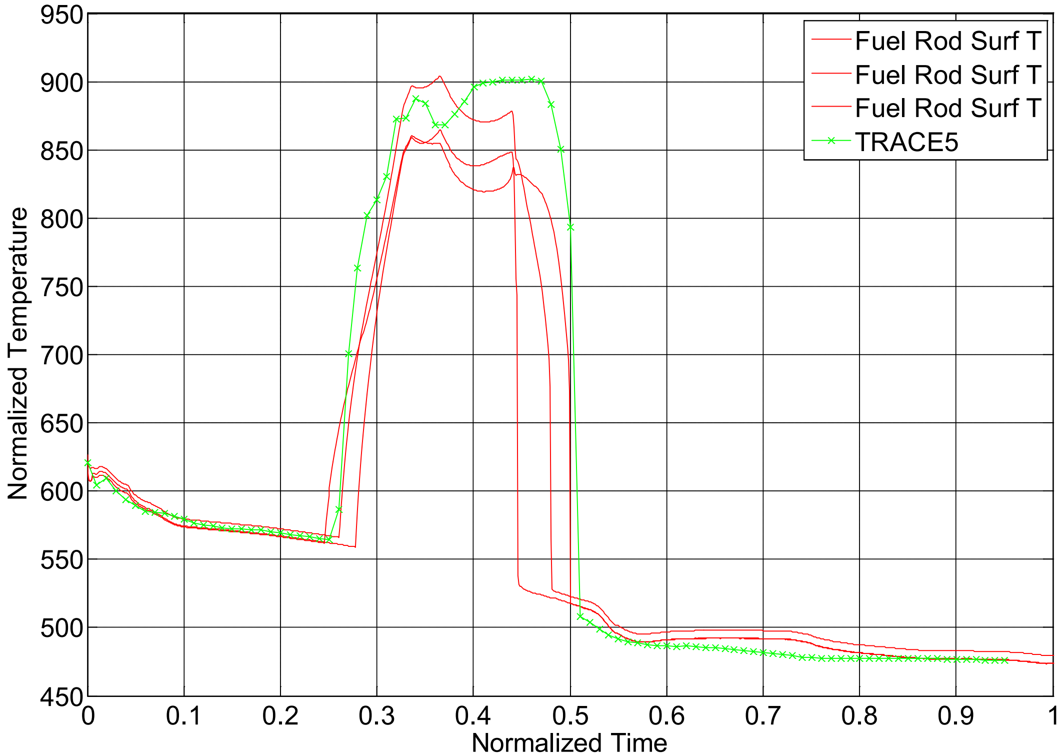

Figure 13 Maximum fuel rod surface temperature.

5.8 <u>Hot and cold legs liquid levels</u>

The following Figures (14, 15, 16 and 17) show the liquid level in hot and cold legs, respectively. Experimentally, liquid level was obtained with a three gamma ray beam densitometer. In the experiment, the collapsed liquid level of both hot legs is almost the same during the Accident Management (AM) action, since the natural circulation phenomenon is produced in both loops. The flow in the hot legs becomes two-phase flow and is horizontally stratified with high liquid velocity after the break. Afterwards it becomes supercritical during two-phase natural circulation due to the high liquid and vapour velocity, causing a drastic drop of the level. At about 0.06 NT, when the two-phase natural circulation ends, the flow in the hot legs becomes subcritical, recovering the liquid level. At 0.18 NT a significant drop of the liquid level is produced, after the inlet plenum of the steam generator is completely empty, and it recovers after the actuation of the accumulator injection system. No transition to supercritical flow is produced in the cold legs since the liquid velocity is not accelerated by the upper plenum. At 0.18 NT the liquid level of the cold legs drops and it temporarily recovers with the injection of the accumulator.

In general, TRACE can reproduce all the tendencies and changes in both hot and cold legs. Most discrepancies are found in the last part of the transient because TRACE does not exactly reproduce the recovering of the liquid level in any of the legs, due to the rapid coolant discharge through the break from the accumulation injection on.

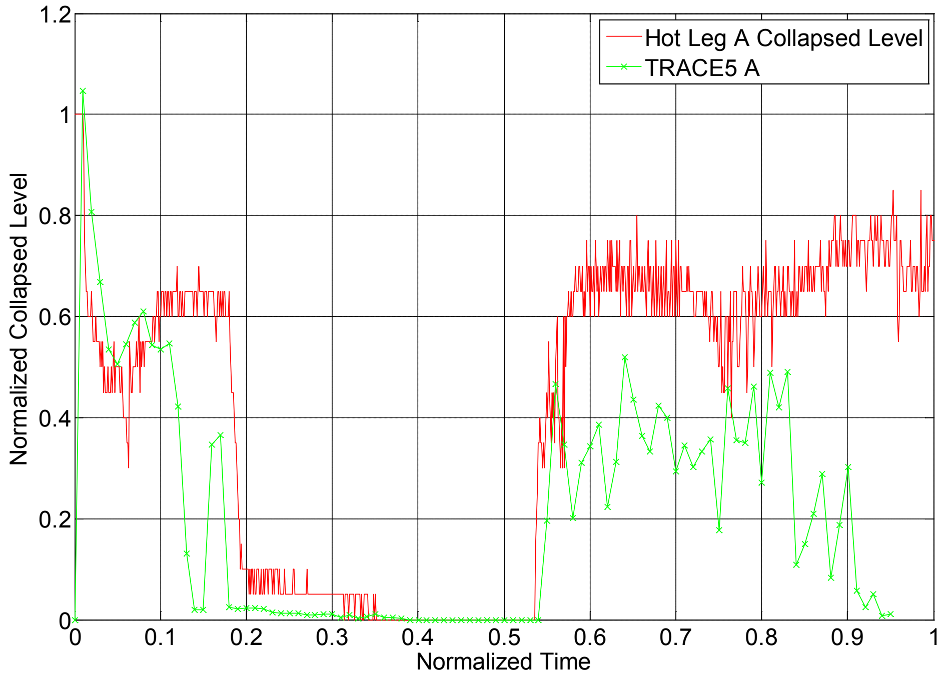

Figure 14 Collapsed liquid level in the hot leg A.

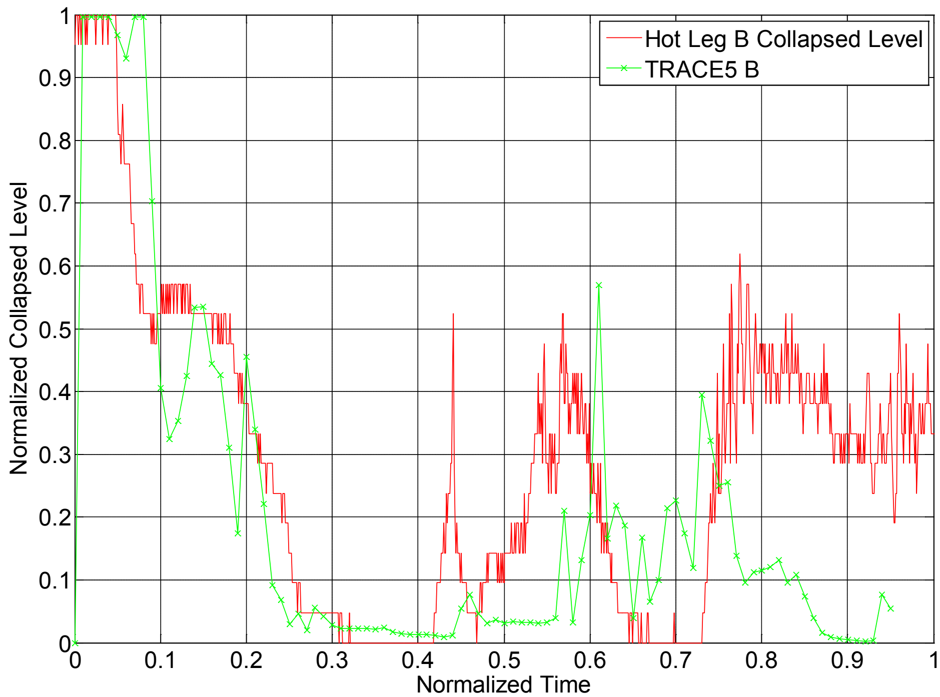

Figure 15 Collapsed liquid level in the hot leg B.

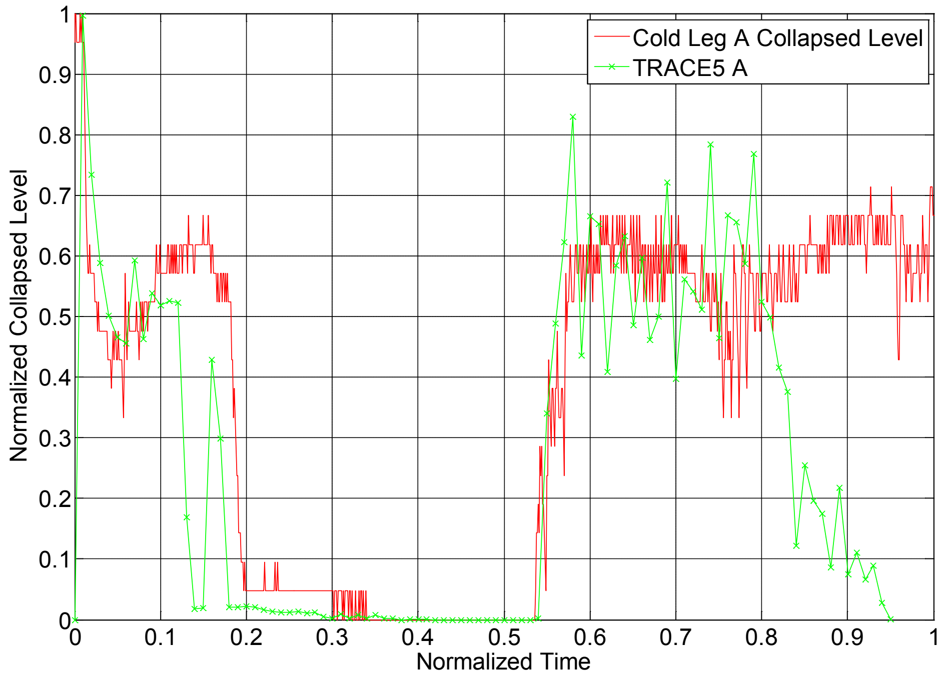

Figure 16 Collapsed liquid level in the cold leg A.

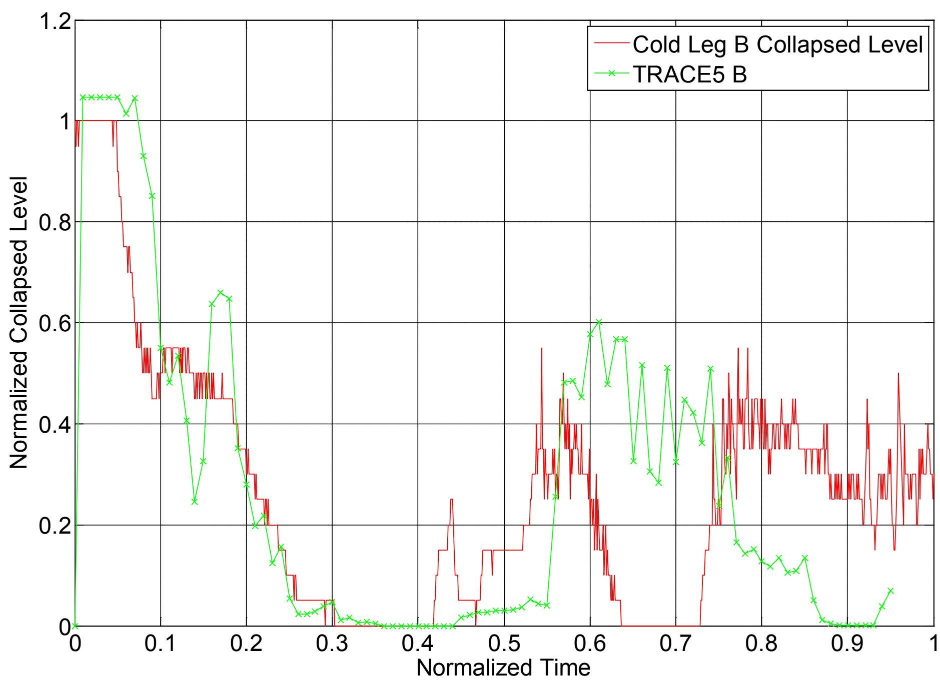

Figure 17 Collapsed liquid level in the cold leg B.

5.9 Froude number calculation

The Froude number is defined as the ratio of inertia to gravity forces in the flow. This ratio may also be interpreted physically as the ratio of the mean flow velocity to the speed of an elementary gravity (surface or disturbance) wave traveling over the water surface. When the Froude number is equal to one, the speed of the surface wave and that of the flow is the same. The flow is in the critical state. When the Froude number is less than one, the flow velocity is smaller than the speed of a disturbance wave traveling on the surface. Flow is considered to be subcritical. Gravitational forces are dominant. The surface wave will propagate upstream and, therefore, flow profiles are calculated in the upstream direction. When the Froude number is greater than one, the flow is supercritical and inertial forces are dominant. The surface wave will not propagate upstream, and flow profiles are calculated in the downstream direction.

Figure 18 shows the evolution of the Froude number in the time interval between 0 and 0.06 NT in both hot legs. Froude number is calculated using the following formula:

$$Fr = \frac{U}{\sqrt{gh}},$$

where U is the liquid velocity (m/s), h the liquid level (m) and g the acceleration due to gravity (m/s^2). Froude number graphic shows that flow is supercritical during the interval 0.0 to 0.007 NT, and between 0.017 to 0.06 NT, both in TRACE5 and in the experiment.

Figure 18 shows that flow in hot legs is supercritical soon after the break, during the interval 0.0 to 0.007 NT, and becomes subcritical during a short period (0.007 to 0.015 NT). Flow in hot legs is again supercritical during two-phase natural circulation (0.015 NT to 0.06 NT), due to the high liquid and vapor velocity by high core power, causing a drastic drop of the level. Finally, flow in hot legs returns to subcritical at 0.06 NT, recovering the hot leg liquid level, when two-phase natural circulation was terminated.

Liquid flow in the hot legs is accelerated by vapor flow and liquid head in the upper plenum when the upper plenum liquid is above the hot leg level. Such water acceleration mechanisms are not observed in the cold legs. No transition to supercritical flow occurs in the cold legs. In general, TRACE5 can reproduce all the tendencies and changes in both hot and cold legs.

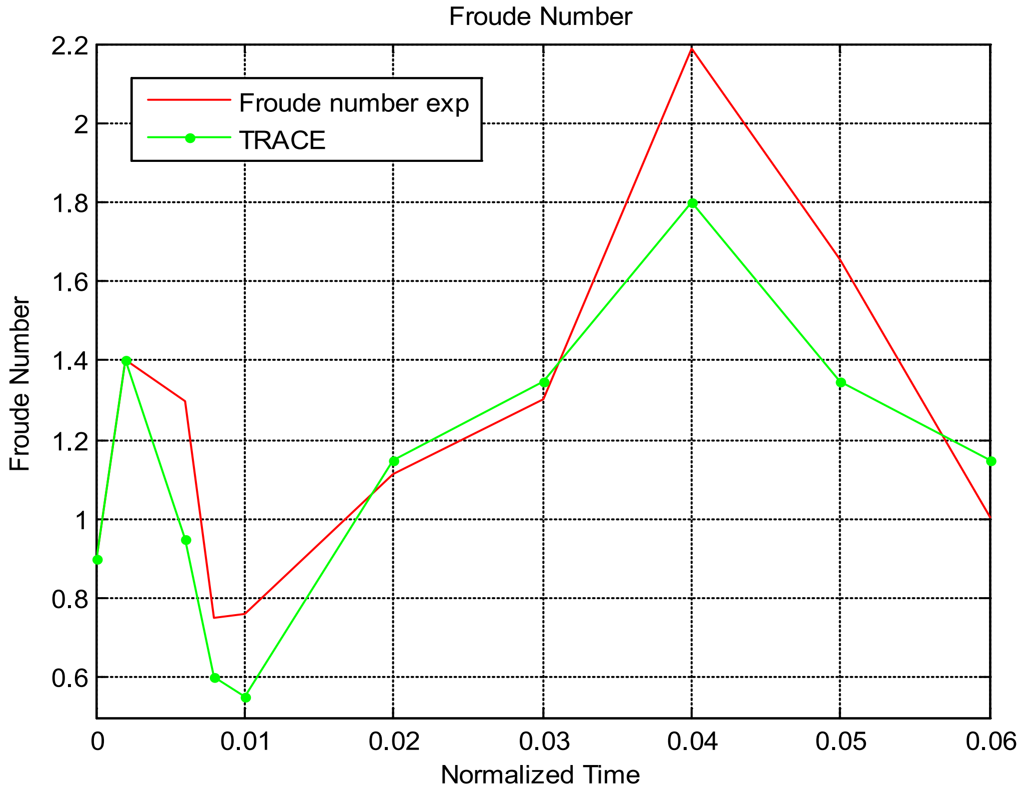

Figure 18 Froude number in the hot leg (0 to 0.06 NT).

5.10 Steam Generator relief valve flow rate

In the TRACE5 model, these valves are represented by means of components VALVE-73 (loop A) and VALVE-75 (loop B). A good agreement has been achieved between TRACE5 and the experiment, as can be observed in Figures 19 and 20. These figures perfectly show the periods corresponding to the RV actuation. It is important to remark that this section only shows the vapour mass flow through the RV.

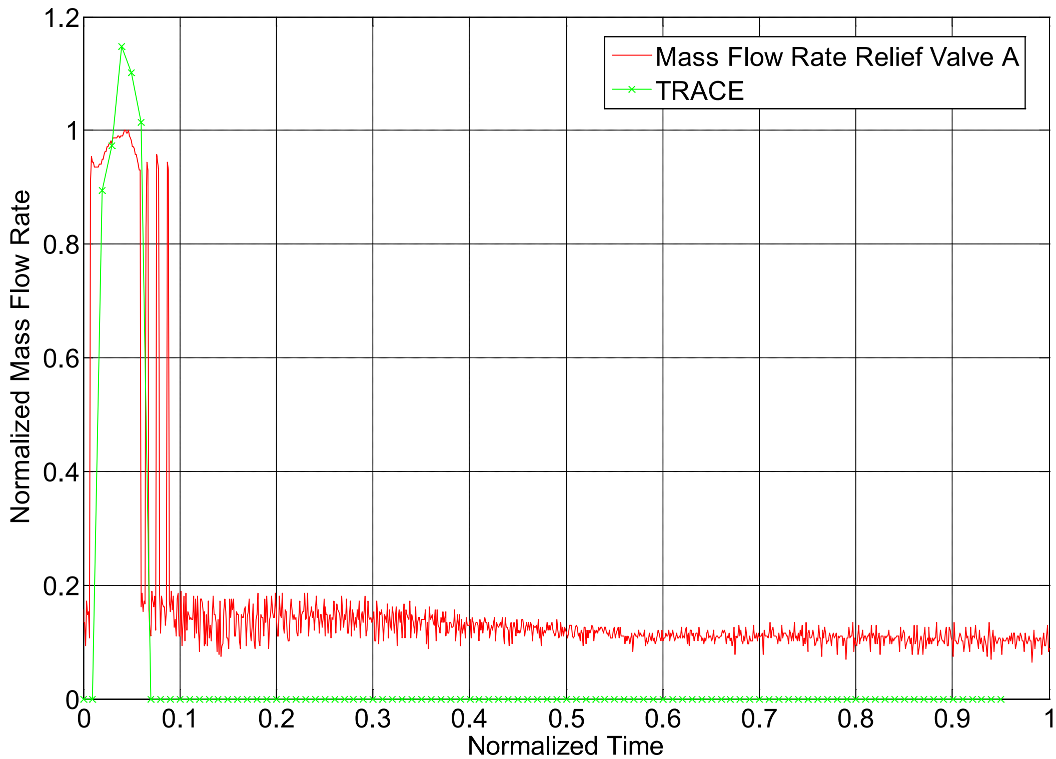

Figure 19 SG A relief valve mass flow.

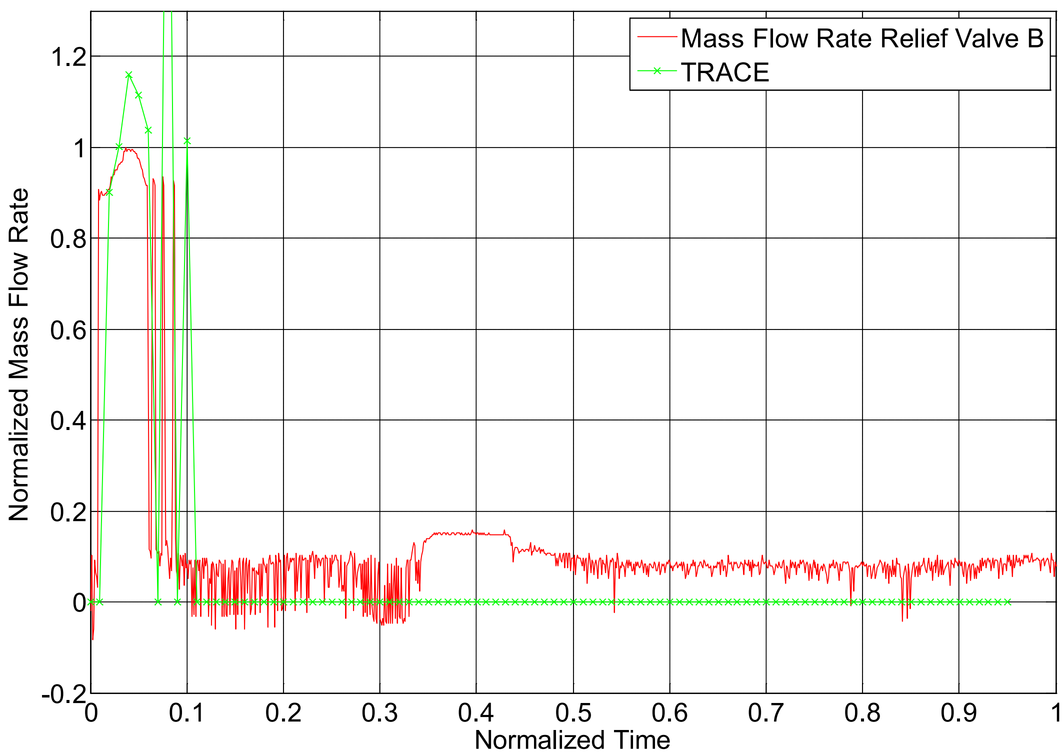

Figure 20 SG B relief valve mass flow.

5.11 Steam Generator Main Steam Isolation Valves mass flow rate

Main Steam Isolation Valves (MSIVs) are automatically closed when the scram signal is generated. This occurs at 0.004 NT in the experiment and at 0.008 NT in the simulation with TRACE5 (Figure 21 and Figure 22). During the rest of the transient, the MSIVs remain closed and therefore the mass flow is zero.

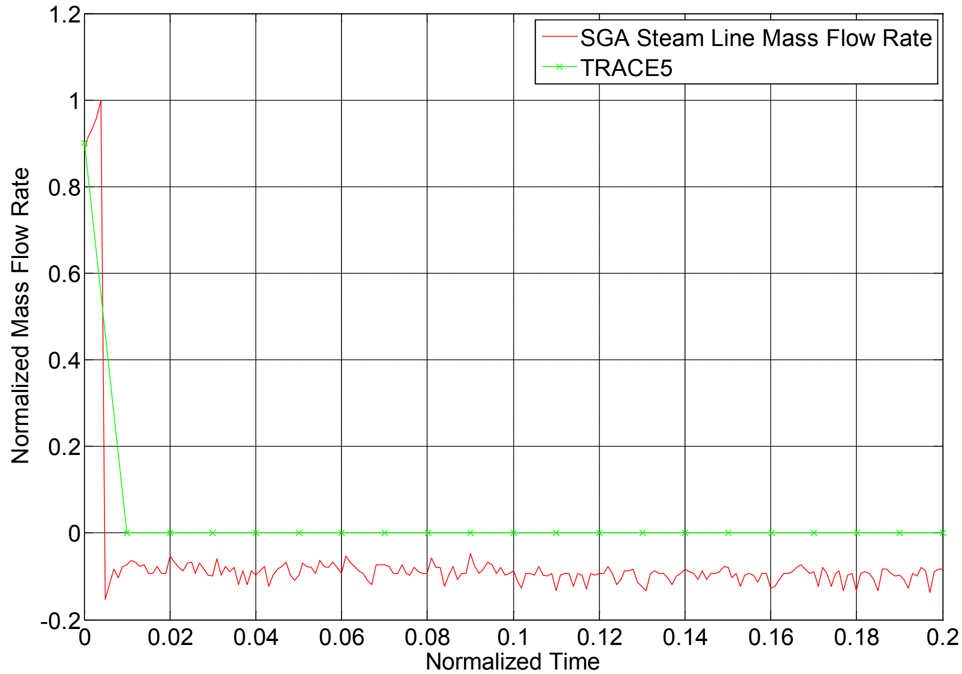

Figure 21 SG A MSIV mass flow rate.

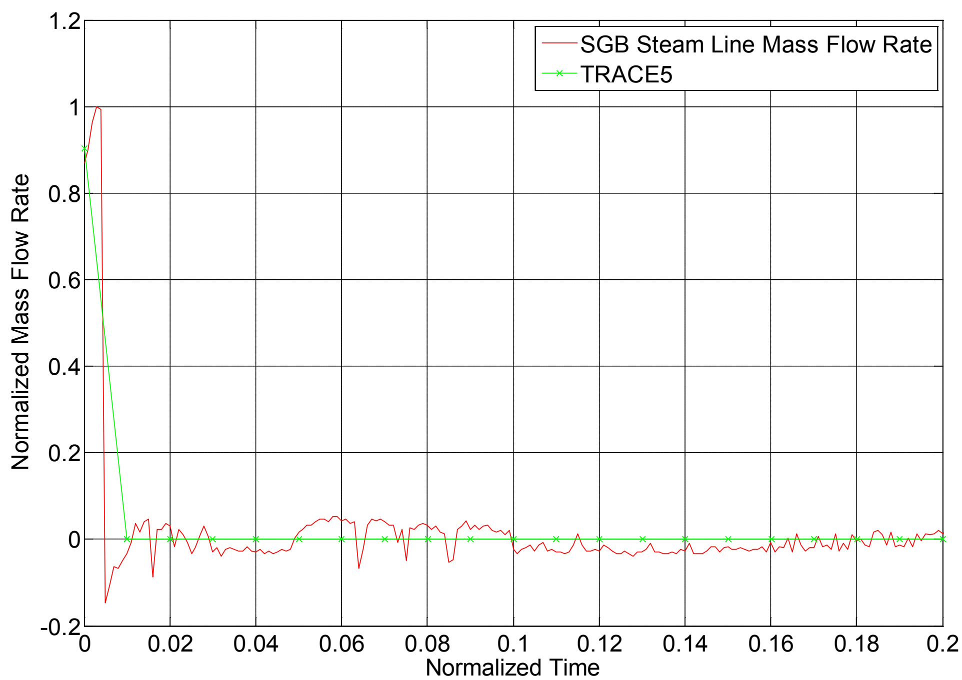

Figure 22 SG B MSIV mass flow rate.

5.12 U-tubes collapsed liquid level

U-tubes have been grouped according to similar lengths. In the TRACE5 simulation and due to the calculation time cost together with the obtained results, a 3-group classification has been adopted. Collapsed liquid levels obtained with TRACE5 are satisfactory, properly reproducing the clearance of the tubes. The collapsed liquid level of the U tubes in the steam generator indicates the beginning of a rapid increase of the void fraction before the core power starts to decrease. The accumulation of liquid in the upflow side and in the inlet plenum takes place during the reflux condensation, probably due to the CCFL in the inlet of the U tubes and in the bottom of the inlet plenum produced by the high vapour velocity caused by the core power. The accumulation of liquid in the upflow side and in the inlet plenum causes a small drop of the core liquid level between 0.06 and 0.1 NT. Results are shown in Figures 23 and 24.

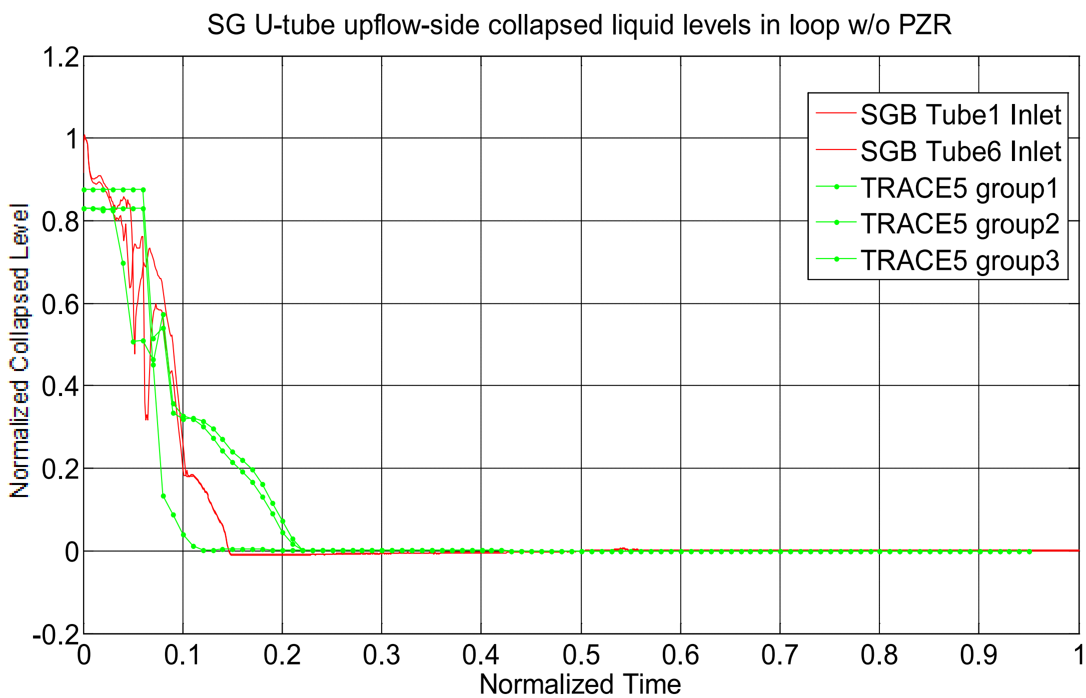

Figure 23 SG U-tube up-flow side collapsed liquid levels in loop B.

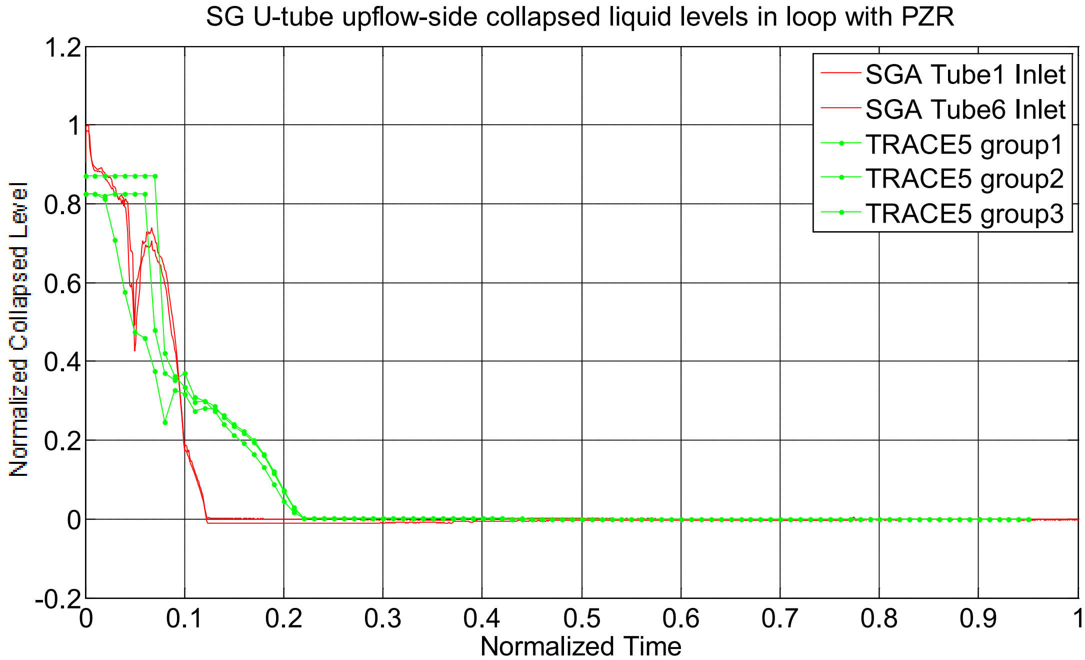

Figure 24 SG U-tube up-flow side collapsed liquid levels in loop A.

5.13 Steam generators secondary-side liquid level

The following Figures (25 and 26) show the collapsed liquid level of the secondary side of the steam generator. Along with the closure of the MSIVs, the main vapour line flow to the turbines is interrupted and the Main Feed Water (MFW) line of both steam generators is closed.

From the generation of the SI signal on, the auxiliary feed water system begins actuating with the purpose of keeping the secondary liquid level of the steam generators.

The stopping of the relief valves is followed by an increase of liquid level from the secondary side of the steam generator due to the vapour condensation caused by the auxiliary feed water. The auxiliary water stops at 0.56 NT when the liquid level in the secondary side of the steam generator reaches over the top of the U tubes. The prediction with TRACE5 is satisfactory.

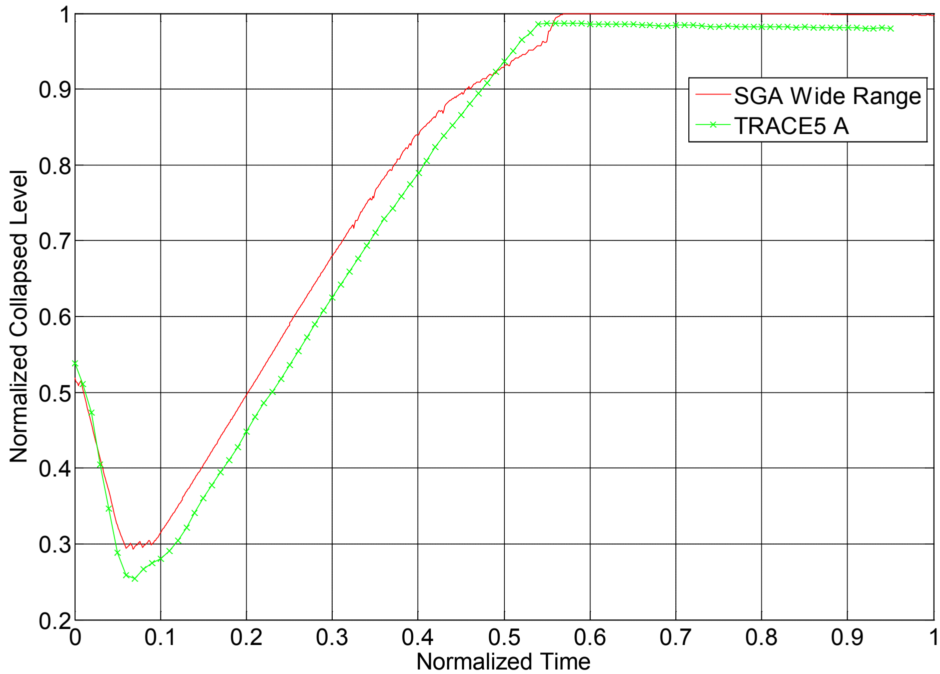

Figure 25 Steam generator A. Secondary-side collapsed liquid level.

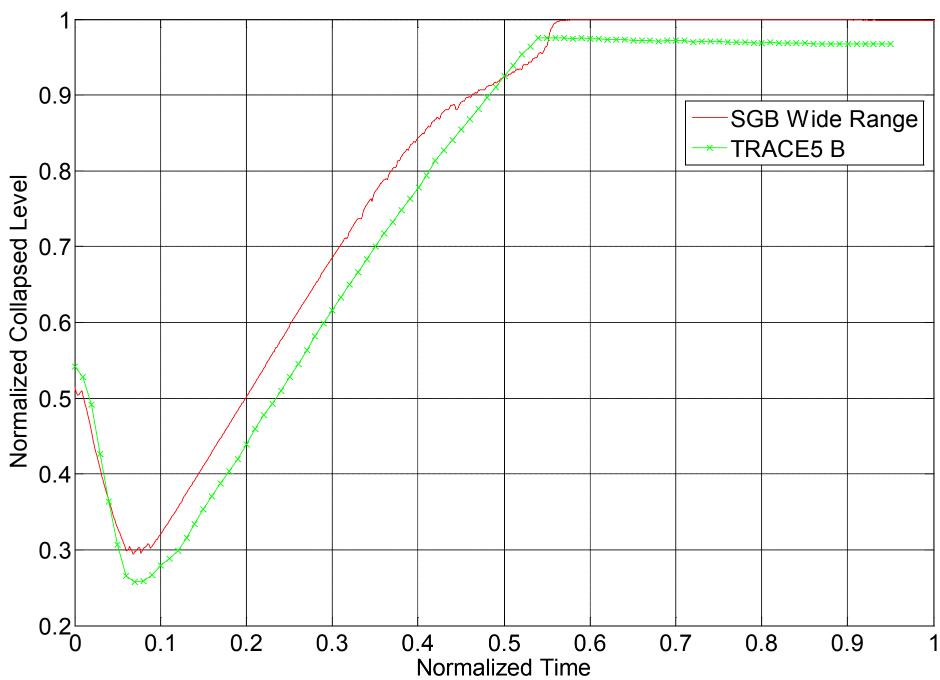

Figure 26 Steam generator B. Secondary-side collapsed liquid level.

5.14 Pressurizer liquid level

Figure 27 shows the pressurizer water level. Liquid level starts to decrease immediately after the break. Experimentally, it becomes completely empty at 0.06 NT. On the other hand, both heaters of the PZR automatically connect to the maximum power after the break, according to the control logic of the depressurization of the transient. Both heaters disconnect at 0.067 NT. The liquid level of the PZR remains empty for the rest of the transient.

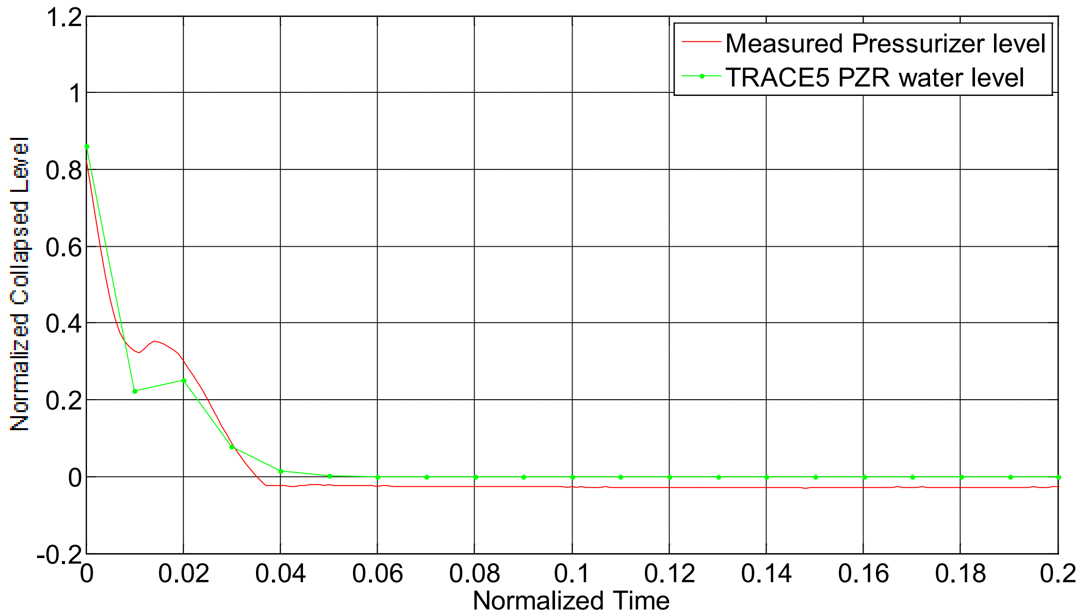

Figure 27 Pressurizer collapsed liquid level (0 to 0.2 NT).

5.15 Hot and cold legs fluid temperatures.

In Figures 28 and 29 it can be seen fluid temperatures measured and simulated in the hot legs A and B, respectively. Fluid temperature in cold leg in loop A is represented in Figure 30.

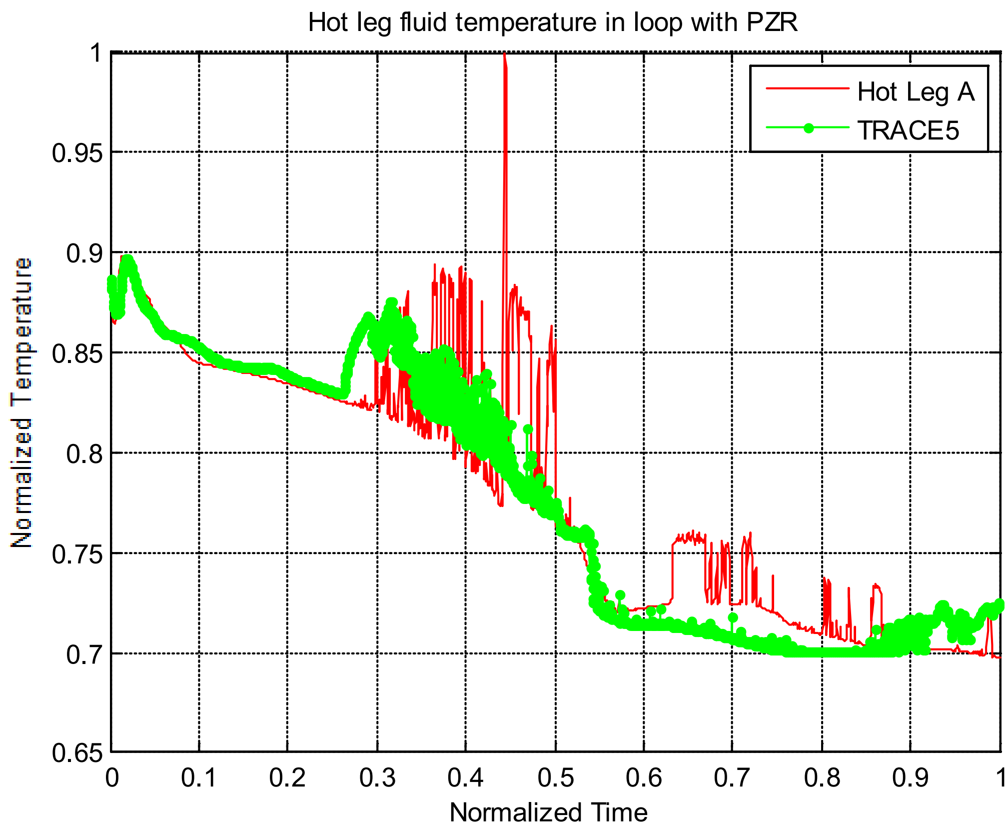

Figure 28 Hot leg fluid temperature in loop A.

Hot leg fluid temperature increases due to the superheated steam coming from the core once the core uncovery has begun.

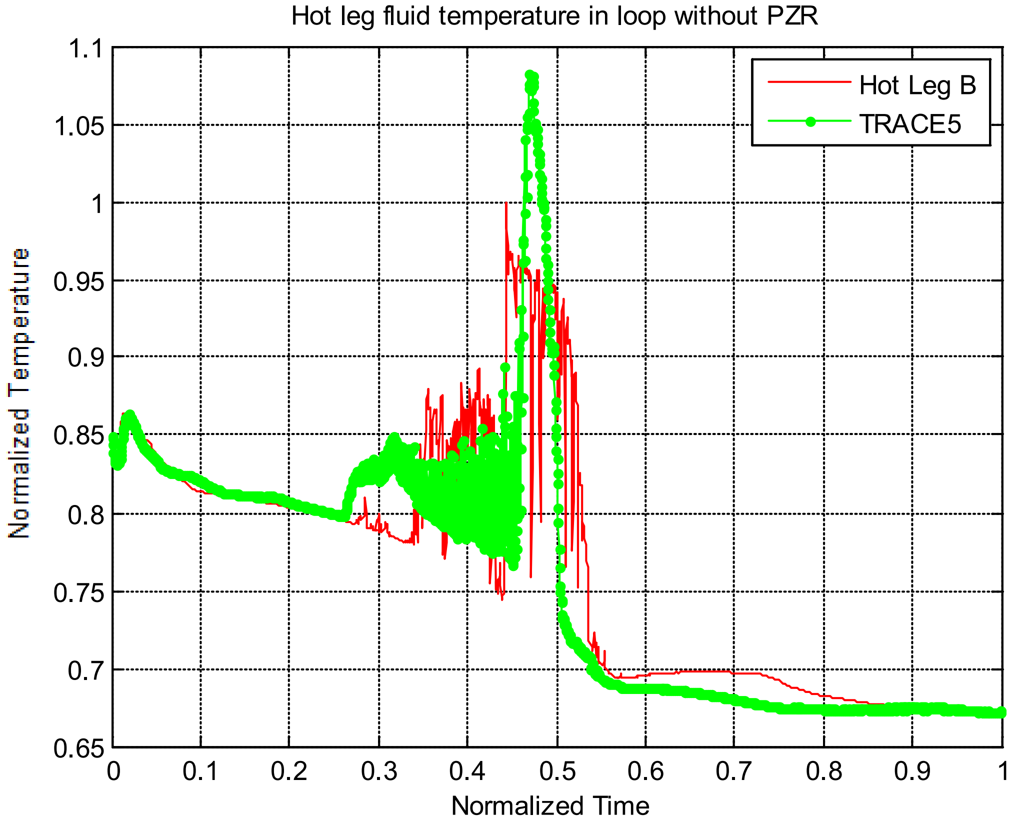

Figure 29 Hot leg fluid temperature in loop B.

Hot leg fluid temperature increases when hot steam flow passes through the crossover leg at about 0.44 NT after the loop seal clearing in the loop without pressurizer. The initiation of the accumulator coolant injection (at 0.4 NT approximately) causes the fluid temperature to decrease at the cold legs.

The main discrepancies between measured and simulated curves are observed in the cold leg loop A after the activation of the core protection system by power control (at 0.5 NT approximately).

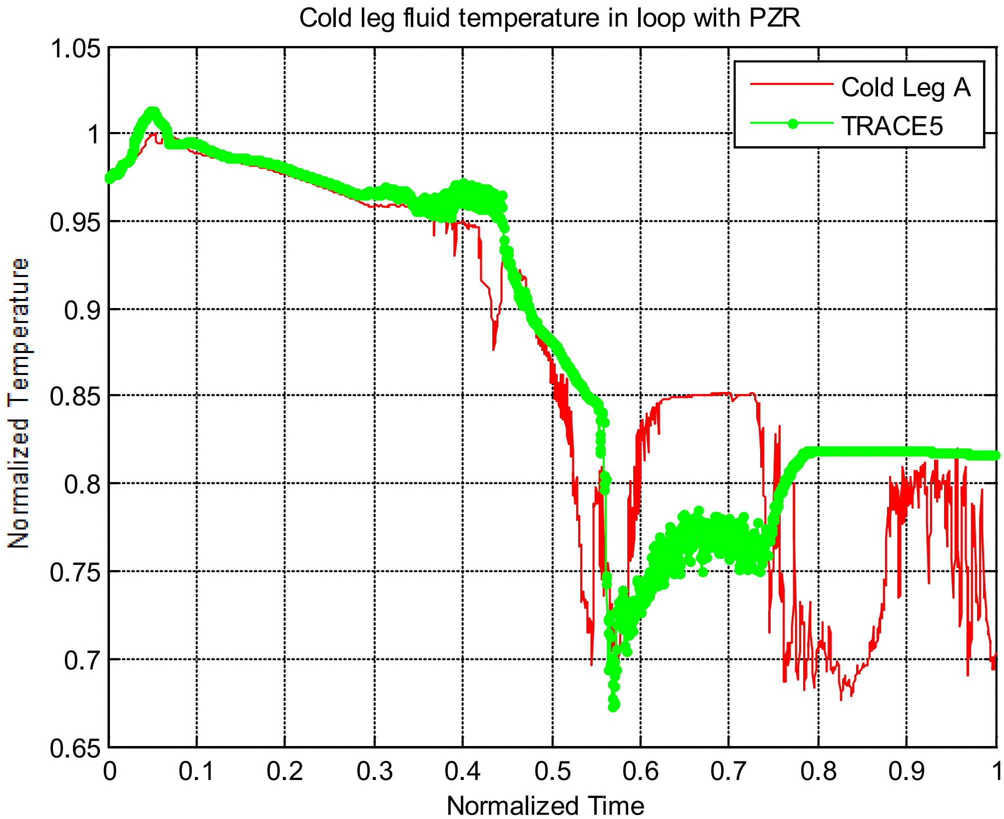

Figure 30 Cold leg fluid temperature in loop A.

6 CONCLUSIONS

The objective of this report was to obtain detailed thermal-hydraulic data in the simulation of the OECD/NEA ROSA Project Test 3-1, high-power natural circulation in a PWR, for the validation of the code TRACE5.

Results show that TRACE5 can successfully reproduce complicated conditions of High Power Natural Circulation, when a break flow in a cold leg is produced. TRACE5 adequately predicts the mass flow and the coolant inventory discharged through the break, and the changes between subcritic and supercritic flow. In fact, flow in hot legs became supercritical during two-phase natural circulation due to high vapor and liquid velocity.

Results with TRACE5 confirm that both hot legs present a similar thermal hydraulic behavior. The only deviations from the experimental results occur during the period when the accumulator actuates. Primary pressure does not increase when coolant is injected from accumulator into the hot leg of the loop without PZR and therefore the coolant is more rapidly discharged through the break. Also hot leg in loop without PZR does not have a sudden increase of mass flow due to this fact, and all hot and cold legs begin to empty before the experimental.

7 REFERENCES

1. H. Y. Jeong, “Prediction of counter-current flow limitation at hot leg pipe during small-break LOCA”, Annals of Nuclear Energy, 29, 571-583 (2002).

2. K. W. Seo, M. H. Kim, M. H. Anderson, M. L. Corradini, “Heat Transfer in a Supercritical Fluid: Classification of Heat Transfer Regimes”, Nuclear Technology, 154, 335-349 (2006).

3. M. L. Corradini, “Transport Phenomena in Supercritical Fluids in GEN-IV Reactor Designs”, Nuclear Technology, 167, 145-156 (2009).

4. B. Freeze, S. Smolentsev, N. Morley, M. Abdou, “Characterization of the Effect of Froude Number on Surface Waves and Heat Transfer in Inclined Turbulent Open Channel Water Flows”, International Journal of Heat and Mass Transfer, 46, 3765-3775 (2003).

5. Thermalhydraulic Safety Research Group, Nuclear Safety Research Center, Final Data Report of ROSA/LSTF Test 3-1 (High Power Natural Circulation Experiment SB-CL-38 in JAEA), Japan Atomic Energy Agency, Japan (2008).

6. T. Takeda, H. Asaka, H. Nakamura, “RELAP5 analysis of OECD/NEA ROSA Project experiment simulating a PWR small break LOCA with high-power natural circulation”, International Conference on the Physics of Reactors “Nuclear Power: A Sustainable Resource”, Casino-Kursaal Conference Center, Interlaken, Switzerland, September 14-19 (2008).

7. The ROSA-V Group, ROSA-V Large Scale Test Facility (LSTF) system description for the 3rd and 4th simulated fuel assemblies, JAERI-Tech, Japan, 2003.

8. Nuclear Regulatory Commission, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, TRACE V5.0. User’s manual. Volume 1: Input Specification, Nuclear Regulatory Commission, U. S (2007).

9. Division of Risk Assessment and Special Projects. Office of Nuclear Regulatory Research. U. S. Nuclear Regulatory Commission, TRACE V5.0. Theory manual. Field Equations, Solution Methods and Physical Models, Nuclear Regulatory Commission, US (2007).

10. Nuclear Regulatory Commission and Applied Programming Technology, Symbolic Nuclear Analysis Package (SNAP) (2007).

NRC FORM 335 (9-2004) NRCMD 3.7 — U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/IA-0413
2. TITLE AND SUBTITLE Assessment of TRACE 5.0 against ROSA test 3-1, Cold Leg SBLOCA	3. DATE REPORT PUBLISHED MONTH: March — YEAR: 2012 4. FIN OR GRANT NUMBER
5. AUTHOR(S) S. Gallardo, V. Abella, G. Verdú and A. Querol	6. TYPE OF REPORT Technical 7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Universitat Politècnica de València
ETSII
Camí de Vera s/n
46021 València, SPAIN

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The purpose of this work is to overview the results provided by the simulation of a cold leg Small Break Loss-Of-Coolant Accident (SBLOCA) under the assumption of total failure of high pressure injection system in the Large Scale Test Facility (LSTF) via the thermal-hydraulic code TRACE5. The work is developed in the frame of OECD/NEA ROSA Project Test 3-1 (SB-CL-38 in JAEA). Test 3-1 simulated a PWR high-power natural circulation due to failure of scram during cold leg SBLOCA with a break size equivalent to 0.1% of the cold leg under assumption of total failure of High Injection System.

A detailed model of the plant and the chronology of events, following these assumptions, has been developed with TRACE5. A comparison of the simulation of the test with the experimental results is provided throughout several graphs. An acceptable general behaviour is observed in the entire transient. In conclusion, this work represents a good contribution for assessment of the predictability of computer codes such as TRACE5.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
Consejo de Seguridad Nuclear (CSN) Thermal-hydraulic CAMP-Spain program TRAC/RELAP Advanced Computational Engine (TRACE) code Small Break Loss-Of-Coolant Accident (SBLOCA) Large Scale Test Facility (LSTF) OECD/NEA ROSA Project Test 3-1 (SB-CL-38 in JAEA).	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

NUREG/IA-0413

Assessment of TRACE 5.0 Against ROSA Test 3-1, Cold Leg SBLOCA

March 2012